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Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

License Renewal Application
Response to Request for Additional Information
(TAC Nos. MC2099 and MC2100)

Reference:
1) Letter from NMC to NRC dated February 25, 2004 (NRC 2004-0016)
2) Letter from NMC to NRC dated August 3, 2004 (NRC 2004-0079)
3) NRC Memorandum from Luis A. Reyes to Chairman Diaz, *et al*, dated May 27, 2004, "Pressurized Thermal Shock Analyses for Renewal of Certain Nuclear Power Plant Operating Licenses"
4) Letter from NMC to NRC dated September 10, 2004 (NRC 2004-0085)
5) NRC Request for Additional Information dated September 23, 2004

In Reference 1, Nuclear Management Company, LLC (NMC), submitted the Point Beach Nuclear Plant (PBNP) Units 1 and 2 License Renewal Application (LRA). In Reference 2, NMC withdrew an associated request for exemption to 10 CFR 50.61 and Appendices G and H to 10 CFR 50. As a result, LRA sections that were presupposed on approval of those exemptions required revising.

Reference 4 provided the revised LRA sections 4.1.2, Identification of Exemptions, 4.2.1, Reactor Vessel Pressurized Thermal Shock, 4.2.2, Reactor Vessel Upper Shelf Energy, 4.2.3, Reactor Vessel Pressure/Temperature Limits, Appendix A, Program and Time Limited Aging Analysis Descriptions, and B2.1.18, Reactor Vessel Surveillance Program. However, the assumptions in Reference 4 were not consistent throughout 4.2.1 and 4.2.2.

In Reference 5, the NRC requested additional information (RAI) regarding their review of Reference 4. The response to RAI question 4.2-1 is contained in Enclosure 1. It provides the same sections as those provided in Reference 4; however, they have been

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revised to incorporate the 2004 fluence projections for the upper shelf energy (USE) Equivalent Margins Analysis.

The response to RAI question 4.2-2 is contained in Section 4.2-1 of Enclosure 1 and as follows. The 2004 fluence projections indicate that the limiting weld will experience this limiting fluence (3.31 X 10^{19} n/cm^2) at 38.1 EFPYs. Assuming a long-term capacity factor of 95%, this fluence would be achieved late in 2017.

This information takes credit for use of the 10 CFR 54.21(c)(1)(iii) option as discussed in Reference 3. Enclosure 1 contains the revised License Renewal Application (LRA) sections and provides a demonstration that the effects of aging on the intended functions of each system, structure, and component will be adequately managed for the period of extended operation. As a result, we have elected to not extend the existing Time Limited Aging Analysis (TLAA) at this time.

To ensure completeness, Enclosure 2 contains BAW-2467NP, Revision 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years", dated October 2004. BAW-2467NP is an analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements, which is being submitted for review and approval in accordance with the requirements of 10 CFR 50, Appendix G, Section IV.A.1.c. Please note that this is a non-proprietary version of the analysis. The proprietary version will be provided upon request.

Should you have any questions concerning this submittal, please contact Mr. James E. Knorr at (920) 755-6863.

Summary of Commitments

To ensure completeness commitments made as part of this submittal are again listed as follows:

1. PBNP will continue to implement the low-low leakage loading fuel management pattern to minimize the limiting weld fluence. In addition, PBNP will continue operation with Hafnium absorber assemblies in service until the resolution of the Unit 2 intermediate-to-lower shell girth weld PTS issue via an alternative analysis methodology.

2. Documentation of a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 Reactor Pressure Vessel intermediate-to-lower shell girth weld will be completed within one -year of receipt of the extended license. Documentation within this time frame will support submittal of any required safety analysis at least three years prior to the time frame that RT_{PTS} for Unit 2 is projected to exceed the screening criteria.

3. If acceptable PTS results cannot be provided prior to EOL with alternate analysis techniques, the PBNP flux reduction program will evaluate the feasibility and practicality of pursuing additional aggressive flux reduction measures prior to EOL, such as the insertion of part length shielded fuel assemblies.

I declare under penalty of perjury that the forgoing is true and correct. Executed on October 25, 2004.

Dennis L. Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

REVISED SECTIONS 4.1.2, 4.2.1, 4.2.2, 4.2.3, APPENDIXES A 15.2.18, 15.4.1, 15.5, AND B 2.1.18 TO THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION

Following are revised sections of the Point Beach Nuclear Plant License Renewal Application which reflect the use of 10 CFR 50.61 as a program to manage the effects of aging on reactor vessel integrity. These sections replace in total those in the February 25, 2004, License Renewal Application and the September 10, 2004 supplemental submittal with the same numbers. Note that the references listed in the text of these sections are the same as those referenced in the original submittal with the addition of Reference 75.

Table 4.1-2 Time limited Aging Analyses

Line number 1, column 5, in Table 4.1-2, Time Limited Aging Analyses (TLAA) is changed to; "*(iii) effects of aging on the intended function will be adequately managed for the period of extended operation.*"

4.1.2 Identification of Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12, and that are in effect, based on time-limited aging analyses, as defined in 10 CFR 54.3.

Active 10 CFR 50.12 exemptions were reviewed to determine whether the exemption was based on a time-limited aging analysis. No TLAA related exemptions granted pursuant to 10 CFR 50.12 were identified.

4.2 Reactor Vessel Irradiation Embrittlement

This group of time-limited aging analyses concerns the effect of irradiation embrittlement on the belt-line regions (adjacent to the reactor core) of the Point Beach Nuclear Plant Units 1 and 2 reactor vessels, and how this mechanism affects analyses that provide operating limits or address regulatory requirements. The calculations discussed in this section use predictions of the cumulative effects on the reactor vessels from irradiation embrittlement. The calculations are based on periodic assessment of the neutron fluence and resultant changes in the reactor vessel material fracture toughness.

The intermediate and lower shells, and welds that join them in the beltline region, of the reactor vessel are fabricated from low alloy steels. These ferritic steels exhibit a ductile-brittle transition that results in fracture toughness property changes as a function of both temperature and irradiation. The material property of particular importance in assessing reactor vessel integrity is fracture toughness, which can be defined as the capability of a material to resist sudden failure caused by crack propagation. Fracture toughness is reduced by neutron irradiation. The measure of fracture toughness of the reactor vessel materials when the reactor vessel is above the brittle fracture/ductile failure transition temperature is referred to as upper-shelf energy. Upper-shelf energy is related to the ability of a material to resist ductile tearing. In addition, the temperature at which the brittle fracture/ductile failure transition occurs increases with increasing radiation. This shift in the transition temperature is referred to as the shift in reference nil ductility transition temperature (RT_{NDT}).

The effect of embrittlement due to neutron bombardment is evaluated for reactor vessel temperatures throughout the range of normal operating values. Heatup and cooldown curves consider normal, relatively slow thermal transients. Pressurized thermal shock transients are characterized by a rapid and significant decrease in reactor coolant temperature with high pressure in the reactor vessel. The high reactor vessel thermal stresses, when combined with the pressure stresses, are assumed to initiate the propagation of a small flaw that is postulated to exist in the reactor vessel beltline. Postulated high pressures could cause propagation of the flaw through the reactor vessel wall.

The first step in addressing the TLAAs associated with neutron embrittlement is the projection of the neutron fluence that the critical vessel locations experience. The Westinghouse Radiation Engineering and Analysis Group performed PBNP reactor vessel fluence projections. The evaluations used the ENDF/B-VI scattering cross-section data set. The calculated fluence projections were determined using methods consistent with Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

These fluence projections were based on historical operational data, and forecasted uprated (1678 MWt) power conditions using a low-low leakage fuel management pattern (L4P) without the presence of Hafnium power suppression absorber rods. The fluence projections performed in 2002 assumed that the units were uprated to 1678 MWt in 2002. The 2002 fluence projections were used as the input basis for the RCS Pressure-Temperature (P-T) Operating Limits required by 10 CFR 50, Appendix G. The fluence projections were revised in 2004 to account for actual unit operational history (including the 1.7 % mini-uprates performed in 2003), and full unit uprates to 1678 MWt in 2008. The 2004 fluence projections were used as the input basis for the Upper Shelf Energy (USE) evaluation required by 10 CFR 50, Appendix G, and the Pressurized Thermal Shock evaluation required by 10 CFR 50.61.

The fluence projections were performed at uprated power conditions to allow for future unit operations at the increased power level.

These fluence projections are bounding and conservative. The analyses for both units have been performed with fluences projected at 53 EFPYs (Effective Full Power Years). The EOEL (End of Extended License) EFPYs for Unit 1, assuming a 95% capacity factor, is forecasted to be 51 EFPYs. The EOEL EFPYs for Unit 2, assuming a 95% capacity factor, is forecasted to be 53 EFPYs.

The results of the calculated neutron fluence values at various locations on the Reactor Pressure Vessel (RPV) are presented in Table 4.2-1.

Table 4.2-1
Summary of the Calculated RPV Neutron Fluence Values at 53 EFPY
(10^{19} n/cm^2, E > 1.0 MeV)

Component Description	Fluence Projection	Surface (i)	1/4T (ii)	3/4T (ii)
PBNP Unit 1: 53 EFPY (End of License Extension)				
Nozzle Belt Forging (122P237)	2002	0.42	0.28	0.13
	2004	0.38	-	-
Inter. Shell Plate (A9811-1)	2002	5.26	3.56	1.63
	2004	5.21	-	-
Lower Shell Plate (C1423-1)	2002	4.79 (iii)	3.24	1.49
	2004	4.83	-	-
Nozzle Belt to Intermediate Shell Circ. Weld (8T1762)	2002	0.42	0.28	0.13
	2004	0.38	-	-
Intermediate Shell Axial Weld - ID 27% (1P0815)	2002	3.44	2.32	1.07
	2004	3.39	-	-
Intermediate Shell Axial Weld - OD 73% (1P0661)	2002	3.44	2.32	1.07
	2004	3.39	-	-
Intermediate to Lower Shell Circ. Weld (71249)	2002	4.91	3.32	1.52
	2004	4.71	-	-
Lower Shell Axial Weld (61782)	2002	3.37	2.28	1.05
	2004	3.25	-	-

Component Description	Fluence Projection	Surface (i)	1/4T (ii)	3/4T (ii)
PBNP Unit 2: 53 EFPY (End of License Extension)				
Nozzle Belt Forging (123V352)	2002	0.55	0.37	0.17
	2004	0.53	-	-
Inter. Shell Forging (123V500)	2002	5.39	3.65	1.67
	2004	5.26	-	-
Lower Shell Forging (122W195)	2002	5.32	3.60	1.65
	2004	5.11	-	-
Nozzle Belt to Intermediate Shell Circ. Weld (21935)	2002	0.55	0.37	0.17
	2004	0.53	-	-
Intermediate to Lower Shell Circ. Weld (72442)	2002	5.09	3.46	1.58
	2004	4.85	-	-

i. These fluence values are the calculated fluence values considering power uprate (1678 MWt) without Hafnium suppression rods.
ii. Neutron attenuation per Reg. Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials (Draft ME 305-4, Proposed Revision 2, published 02/1986), Rev. 2."
iii. 4.79 is in error due to a calculational summary transpositional anomaly. The correct value is 5.06. There is no affect on the generation of the PT curves.

In addition to the plant specific neutron exposure calculations, dosimetry sets from three (3) in-vessel and twenty (20) ex-vessel sensor sets irradiated at Unit 1 and four (4) in-vessel and twenty (20) ex-vessel sensor sets irradiated at Unit 2 were also re-analyzed using dosimetry evaluation methodologies that follow the guidance provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The results of these dosimetry re-evaluations were then used to validate the calculational models that were applied in the plant specific neutron transport analysis of the PBNP RPVs.

The welds in the reactor vessel are basically the same material as the parts being joined and may be considered to be included in the preceding discussions. The chemistry differences between weld metal and base metal affect the material properties that are degraded by embrittlement; therefore, the welds are evaluated separately when considering the aforementioned aging effect. The fracture toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan. The beltline material properties of the Point Beach

Unit 1 and 2 reactor vessels are presented in Table 4.2-2. The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date.

Table 4.2-2
Summary of the Best Estimate Cu and Ni Weight Percent, Initial RT_{NDT} Values and Chemistry Factor values for PBNP Units 1 and 2 Reactor Vessel Materials

Material Description	wt.% Cu	wt.% Ni	Initial RT_{NDT}	CF
PBNP Unit 1				
Nozzle Belt Forging (122P237)	0.11	0.82	50°F	77°F
Inter. Shell Plate (A9811-1)	0.20	0.06	1°F	88°F 79.3°F(i)
Lower Shell Plate (C1423-1)	0.12	0.07	1°F	55.3°F 35.8°F(i)
Nozzle Belt to Intermediate Shell Circ. Weld (8T1762)	0.19	0.57	-5°F	152.4°F
Intermediate Shell Axial Weld - ID 27% (1P0815)	0.17	0.52	-5°F	138.2°F
Intermediate Shell Axial Weld - OD 73% (1P0661)	0.17	0.64	-5°F	157.6°F
Intermediate to Lower Shell Girth Weld (71249)	0.23	0.59	10°F	167.6°F
Lower Shell Axial Weld (61782)	0.23	0.52	-5°F	157.4°F 163.3°F(i)
PBNP Unit 2				
Nozzle Belt Forging (123V352)	0.11	0.73	40°F	76°F
Inter. Shell Forging (123V500)	0.09	0.70	40°F	58°F
Lower Shell Forging (122W195)	0.05	0.72	40°F	31°F 43°F (i)
Nozzle Belt to Intermediate Shell Girth Weld (21935)	0.18	0.70	-56°F(ii)	170°F
Intermediate to Lower Shell Circ. Weld (72442)	0.26	0.60	-5°F	180°F

i. Per Regulatory Guide 1.99, Rev. 2, Position 2.1.
ii. Generic Value of RT_{NDT}.

The calculated fluences and RPV material properties noted above were used in the TLAA calculations associated with RPV neutron embrittlement.

In addition, changes in RPV material properties are verified through surveillance specimen irradiation and testing.

Westinghouse Electric Company developed the original surveillance program for the PBNP Units 1 and 2 RPVs. Although the original program was in accordance with ASTM E 185-66, subsequent testing has followed the latest version of ASTM E 185 that was approved by the NRC, through ASTM E 185-82. A description of the surveillance program and the pre-irradiation mechanical properties of the reactor vessel materials are presented in WCAP-7513 for Unit 1 (Reference 57), and WCAP-7712 for Unit 2 (Reference 58). The original PBNP surveillance program consisted of six surveillance capsules in each Unit attached to the outside of the reactor vessel internals thermal shield. Each capsule contained mechanical specimens, dosimetry, and thermal monitors. The mechanical specimens were fabricated from material representative of the PBNP RPVs.

To date, four surveillance capsules have been removed and tested from each Unit's RPV. One of the standby capsules has also been removed from each Unit's RPV and is being stored at Point Beach. The final, originally installed standby capsule, remains in each PBNP RPV.

The surveillance materials in the capsules of PBNP Units 1 and 2, and other early plant specific Reactor Vessel Surveillance Programs (RVSPs) were not selected in accordance with ASTM E 185-82. Hence, the materials monitored by the RVSPs are not always the materials judged in 10 CFR 50 Appendix H, to most likely be the controlling beltline region materials with regard to irradiation embrittlement for the RPV for which the RVSP was designed. Consequently, the applicability of the data generated in the plant specific RVSP is limited. However, by combining the data developed from several RVSPs, it is possible to use data developed in a given RVSP for application at a different RPV, and also practical to develop a database to predict irradiation behavior of those welds for which there is no specific data. This does not preclude plant specific characterization should sufficient credible surveillance data become available.

Although the PBNP Units 1 and 2 specific surveillance program capsules contained mechanical specimens representative of the materials of the PBNP RPVs, the capsules did not contain materials representative of the PBNP RPVs limiting welds. Since the actual heat of the limiting weld metal for either of the PBNP Units RPV is not in the respective Unit's surveillance program, participation in the B&W Owners Group (B&WOG) Master Integrated Reactor Vessel Surveillance Program (MIRVP) (Reference 59) allows access to irradiated surveillance data of the PBNP limiting RPV welds.

The MIRVP combines 16 separate plant specific RVSPs and provides for sharing of irradiation sites. It addresses requirements for acquiring irradiation data and the need to improve the quality and quantity of fracture toughness data to support operation of the participating plants.

The MIRVP correlates data from both power reactor surveillance monitoring and test reactor research programs. The principal sources of information are the power reactor surveillance efforts; which consists of three parts. The first part is the continuation of the plant-specific RVSPs that monitor the irradiation damage to selected materials, as originally planned and licensed. These capsules contain samples of weld metal, plate, forging, and heat-affected zone (HAZ) material from the vessel beltline, and neutron dosimetry and thermal monitors. This part of the program will continue to monitor the long-term effects of neutron irradiation on the reactor vessel materials.

The second part of the program consists of a series of specially designed supplementary weld metal surveillance capsules (SUPCAPS) to study the effects of irradiation on a number of weld metals. The welds were selected because they were anticipated to be highly sensitive to irradiation damage because of their chemical composition and low initial Charpy upper shelf energies. These capsules differ from regular plant-specific RVSP capsules in that they include the necessary specimens to obtain fracture toughness properties of individual weld metals. The capsules are located in the same irradiation holder tubes as the regular plant-specific capsules at Crystal River-3 and Davis Besse.

The third part of the MIRVP consists of higher fluence supplementary weld metal surveillance capsules (HUPCAPS) to obtain irradiated weld metal data (primarily fracture toughness properties) to satisfy the requirements 10 CFR 50, Appendices G and H, and 10 CFR 50.61 for the current license and license renewal of the plants in the MIRVP. Additional objectives are to (1) provide a capsule of Westinghouse design for correlation of irradiation data in the Westinghouse neutronic environment with the B&W 177-FA environment; (2) provide irradiation of reconstituted specimens to accelerate data gathering; and, (3) provide definitive information on the annealing response of this family of materials.

The PBNP Unit 1 remaining original plant-specific RVSP capsule contains SA-1263 weld material that is a surrogate for SA-1585 and SA-1650, but is not relevant to the PBNP Unit 1 RPV limiting weld materials. The limiting beltline welds for PBNP Unit 1 are SA-847 and SA-1101. SA-847 is covered by surrogate materials SA-1036 in Ginna, and SA-1135 in SUPCAPS. The SA-1101 material is in the Turkey Point Unit 3 RVSP and SA-1094, a surrogate for SA-1101, is in the Turkey Point Unit 4 RVSP. SA-1263 benefits Surry Units 1 and 2 and Oconee Unit 1. However, it is covered in the SUPCAPS and HUPCAPS and no additional data is required for this weld material.

These MIRVP capsules contain several Charpy V-notch and compact fracture toughness specimens of the WF-847 & SA-1101 weld material. EOEL data currently exists for the SA-847 surrogate material SA-1036. Additionally, Capsule A2 will be removed at a target EOEL fluence of 3.7×10^{19} n/cm2. This capsule will be removed in approximately 2008 and will be used in EOEL evaluations of the PBNP Unit 1 SA-847 material.

In addition, a new PBNP surveillance capsule has been installed in PBNP Unit 2 for the purpose of obtaining relevant fracture toughness data at the EOEL fluence. The new PBNP Unit 2 surveillance capsule contains surveillance specimens that will be used to determine the fracture toughness of the PBNP Unit 1 weld material SA-1101. When removed and tested, this surveillance capsule will provide EOEL data for the SA-1101 weld material. The supplemental surveillance capsule for PBNP Unit 2 was installed following Cycle 25. Details regarding the specific contents of the supplemental capsule may be found in WCAP-15856.

The PBNP Unit 2 remaining original plant-specific RVSP capsule contains WF-193 weld material that is a surrogate for WF-112 and WF-154, but is not relevant to the PBNP Unit 2 RPV limiting weld materials. The limiting beltline weld for PBNP Unit 2 is SA-1484. HUPCAP A3 provided data on SA-1484 weld material with a fluence of 1.7×10^{19} n/cm^2. The WF-67 weld was produced using the same weld wire (heat 72442) as the SA-1484 weld and is well characterized in the SUPCAPS and HUPCAPS.

These MIRVP capsules contain several Charpy V-notch and compact fracture toughness specimens of the WF-67 weld material. Two of these capsules have a target fluence of 3.0×10^{19} n/cm^2, which is approximately the projected EOL fluence for PBNP Unit 2. Capsule A1 was scheduled for removal from Davis Besse in 2008. Capsule A4 in Crystal River 3 should be available at about this same time depending upon the actual operating schedule. The exact status for capsule A1 will depend upon a revised operation schedule at Davis Besse. Capsule L2 in Davis Besse has a lower target fluence and thus has little relevance for the PBNP Unit 2 vessel. When any or all of these specimens are tested, the new results will be integrated with the existing data to further assess RPV integrity.

In addition, as stated previously, a new PBNP Unit 2 surveillance capsule has been installed in PBNP Unit 2 for the purpose of obtaining relevant fracture toughness data at the EOEL fluence. The new Unit 2 surveillance capsule contains surveillance specimens that will be used to determine the fracture toughness of the PBNP Unit 2 weld metal heat 72442, as well as, weld and plate materials from PBNP Unit 1 RPV and a weld for the Davis Besse RPV. The supplemental surveillance capsule for PBNP Unit 2 was installed following Cycle 25. Details regarding the specific contents of the supplemental capsule may be found in WCAP-15856 (Reference 60).

The target fluence for the PBNP Unit 2 supplemental surveillance capsule will correspond to the peak reactor vessel fluence at EOEL for the limiting weld metal. Surveillance data obtained from this capsule will provide fracture toughness measurements for the limiting weld metal at EOEL fluence. The EOEL peak fluence estimate for the PBNP Unit 2 circumferential weld is 5.085×10^{19} n/cm^2 and considers the affects of hafnium removal and power uprate. The resulting data will provide direct evidence to demonstrate adequate reactor vessel fracture toughness throughout the license renewal term.

Based on the fluence lead factor for the capsule irradiation location, the supplemental surveillance capsule should be removed and tested at just over 38 EFPY (representing an EOEL fluence for the capsule materials).

The PBNP Units 1 and 2 specific reactor vessel surveillance program coupled with participation in the B&WOG MIRVP meets the requirements of 10 CFR 50 Appendix H.

There are three distinct time-limited aging analyses associated with Reactor Vessel Irradiation Embrittlement:

- Pressurized Thermal Shock evaluation required by 10 CFR 50.61.
- Upper Shelf Energy (USE) evaluation required by 10 CFR 50, Appendix G.
- RCS Pressure-Temperature (P-T) Operating Limits required by 10 CFR 50, Appendix G.

Each of these analyses is discussed separately below.

4.2.1 Reactor Vessel Pressurized Thermal Shock

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the NRC issued a formal rule on PTS, 10 CFR 50.61. It established the screening criteria for pressurized water reactor (PWR) vessel embrittlement as measured by the reference temperature termed RT_{PTS}. Screening criteria were set corresponding to EOL plant operation for beltline axial weld seams, forgings, and plates at 270°F, and at 300°F for beltline circumferential weld seams. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with these criteria through the end of plant operation.

The NRC amended its regulations for PWR plants to change the procedure for calculating radiation embrittlement RT_{PTS} values. The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991, and later updated on December 19, 1995 with an effective date of July 29, 1996.

These amendments made the procedure for calculating RT_{PTS} values consistent with the method given in Regulatory Guide 1.99, Revision 2.

Pressurized thermal shock analyses were performed for the PBNP Unit 1 RPV by Westinghouse using the 2004 fluence projections. These analyses were performed at full uprated power conditions (1678 MWt), without Hafnium absorber rods, for a 60-year operating period. RT_{PTS} values were calculated for the inside surface of the beltline region materials for the Unit 1 RPV using Charpy based fracture toughness evaluations in accordance with the methods of 10 CFR 50.61. The values are summarized in Table 4.2.1-1.

Table 4.2.1-1
Summary of Unit 1 Calculated RT_{PTS} Values RPV Inside Surface, 53 EFPY, 1678 MWt, Without Hafnium, Charpy Based Methodology

Component Description	Fluence Factor	ΔRT_{PTS}(°F)	RT_{PTS}(°F)
Nozzle Belt Forging (122P237)	0.73	56.21	140
Inter. Shell Plate (A9811-1)	1.41	124.08	189
		111.81 (i)	169 (i)
Lower Shell Plate (C1423-1)	1.40	77.42	142
		50.12 (i)	107 (i)
Nozzle Belt to Intermediate Shell Circ. Weld (8T1762)	0.73	111.25	174
Intermediate Shell Axial Weld - ID 27% (1P0815)	1.32	182.42	245
Intermediate to Lower Shell Circ. Weld (71249)	1.39	232.96	299
Lower Shell Axial Weld (61782)	1.31	158.71	222
		213.92 (i)	257 (i)

i. Per Regulatory Guide 1.99, Rev. 2, Position 2.1.

The RT_{PTS} values for the Unit 1 Reactor Pressure Vessel beltline region materials at the EOEL were calculated to be lower than the applicable screening criteria values established in 10 CFR 50.61. The analyses associated with PTS for the Unit 1 RPV have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

PTS analyses were performed for the PBNP Unit 2 RPV by Westinghouse using the 2004 fluence projections. These analyses were performed at full uprated power conditions (1678 MWt), without Hafnium absorber rods, for a 60-year operating period.

RT_{PTS} values were calculated for the inside surface of the beltline region materials for the Unit 2 RPV using Charpy based fracture toughness evaluations in accordance with the methods of 10 CFR 50.61. The values are summarized in Table 4.2.1-2.

Table 4.2.1-2
Summary of Unit 2 Calculated RT_{PTS} Values RPV Inside Surface, 53 EFPY, 1678 MWt, Without Hafnium - Charpy Based Methodology

Component Description	Fluence Factor	ΔRT_{PTS}(°F)	RT_{PTS}(°F)
Nozzle Belt Forging (123V352)	0.82	62.32	136
Inter. Shell Forging (123V500)	1.41	81.78	156
Lower Shell Forging (122W195)	1.41	43.71	118
		60.63 (i)	118 (i)
Nozzle Belt to Intermediate Shell Circ. Weld (21935)	0.82	139.40	149
Intermediate to Lower Shell Circ. Weld (72442)	1.40	252.00	316

i. Per Regulatory Guide 1.99, Rev. 2, Position 2.1.

The RT_{PTS} values for the Unit 2 Reactor Pressure Vessel beltline region materials at the end of the extended operating period were calculated to be lower than the applicable screening criteria values established in 10 CFR 50.61, with the exception of the intermediate to lower shell circumferential weld. The intermediate to lower shell circumferential weld is the limiting Unit 2 Reactor Pressure Vessel weld.

It should be noted that all the RT_{PTS} values are lower than the screening criteria values established in 10 CFR 50.61 for the current license period.

As shown in the above Table, the EOEL fluence yields an RT_{PTS} value of 316°F when using Charpy based methods for the limiting weld for a power uprate to 1678.0 MWt and removal of the Hafnium power suppression assemblies. The screening criteria established in 10 CFR 50.61 (300°F) will be exceeded for the limiting Unit 2 intermediate to lower shell girth weld at a neutron fluence of 3.31×10^{19} n/cm^2. The 2004 fluence projections indicate that the limiting weld will experience this limiting fluence at 38.1 EFPYs. Assuming a long-term capacity factor of 95 %, this fluence would be achieved late in 2017.

10 CFR 50.61(b)(3) states "For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS

screening criterion set forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated approval by the Director, Office of Nuclear Reactor Regulation, of detailed plant-specific analyses, submitted to demonstrate acceptable risk with RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques."

PBNP Flux Reduction Program

The PBNP Reactor Vessel Surveillance Program will include a flux reduction program to manage the Unit 2 RPV intermediate-to-lower shell girth weld PTS issue for the period of extended operation in accordance with 10 CFR 50.61(b)(3).

PBNP RPV embrittlement was recognized as a potential issue early in plant life. A flux reduction program was implemented for both PBNP Units. The initial flux reduction efforts incorporated a "low leakage loading pattern" (L3P) fuel management plan. The low leakage loading pattern fuel management plan was implemented in 1980 for both PBNP Units 1 & 2. Incorporation of the low leakage loading pattern fuel management plan provided a vessel flux reduction of approximately 25 to 30 percent.

PBNP RPV embrittlement was recognized as a potentially limiting issue in early plant life extension studies. Westinghouse performed a Reactor Vessel Flux Reduction Evaluation to identify the best means of reducing the rate of neutron embrittlement of the reactor vessels. Neutron flux reduction goals were established to maintain RPV weld properties within perceived regulatory limits through EOEL. A basic criteria imposed on the flux reduction program was that any flux reduction measures would not adversely affect plant reliability and capacity.

Fourteen (14) possible fuel management techniques, and lower internals redesign and replacement were considered. Redesign and replacement of the lower internals package was not found to be practical. Two (2) of the evaluated fuel management techniques met the neutron flux reduction goals. These 2 fuel management techniques included a low-low leakage loading pattern (L4P) with Hafnium neutron absorber assemblies in the guide tubes of peripheral fuel assemblies (12 locations per Unit), and a low-low leakage loading pattern with modified fuel assemblies, with stainless steel rods - in place of fuel rods, on the outboard side of peripheral fuel assemblies (12 locations per Unit). The technique consisting of a low-low leakage loading pattern with modified fuel assemblies was not considered practical, and was rejected based on cost, reduced core thermal design margins, increased possibility of fuel assembly damage, and an increased amount of material that needed to be disposed of as high-level nuclear waste.

Implementation of a flux reduction program was determined to be prudent, and the flux reduction technique of a low-low leakage loading pattern (L4P) with hafnium neutron absorber assemblies was selected. The transition to a low-low leakage loading pattern, and installation of the hafnium neutron absorber assemblies was performed during the refueling outages in April 1989 for Unit 1 and October 1989 for Unit 2.

To verify the analytical flux reduction predictions, ex-vessel neutron dosimetry sets were installed in the reactor cavity annulus.

The transition to a L4P plus Hafnium flux reduction program achieved an approximate flux reduction factor of 1.5 to 2, and was initially forecasted to achieve the goals for EOEL. Subsequently, incorporation of extended cycles (> 12 months) increased projected EOEL fluences through increased flux rates and an increase in unit capacity factor. In addition, changes occurred in the definition of limiting weld material properties and PTS calculational constraints. These issues resulted in the forecasted Unit 2 RPV limiting weld PTS value exceeding the acceptance criteria at EOEL.

The PTS screening criteria will not be met at EOEL for the limiting Unit 2 weld even if the power level is held at the current CLB level (1540 MWt), with the continued presence of Hafnium neutron absorber assemblies installed in the peripheral fuel assemblies.

Flux reduction measures that limit plant capacity (power level or capacity factor) are not considered reasonable or practical options. Maximizing power output is an important factor in ensuring that PBNP continues to provide a cost competitive product.

PBNP will continue to implement the low-low leakage loading fuel management pattern to minimize the limiting weld fluence. In addition, PBNP will continue operation with Hafnium absorber assemblies in service until the resolution of the Unit 2 intermediate-to-lower shell girth weld PTS issue via an alternate analysis methodology.

Continued funding of Hafnium neutron absorber assemblies over the long term is not reasonable since the flux reduction provided by these devices does not prevent the limiting weld PTS value from exceeding the acceptance criteria.

<u>Improved Analysis Technique Option</u>

The current ASME Code reference toughness methodology is generally very conservative. The Master Curve method of using measured, small-specimen, fracture toughness properties to define a new indexing temperature, with a statistically derived lower bound tolerance curve, more accurately describes the fracture toughness behavior of RPV materials.

The Master Curve fracture toughness approach has precedence with the Nuclear Regulatory Commission (NRC) as evidenced in the Safety Evaluation (SE) issued for the Kewaunee Nuclear Power Plant approving the application of a Master Curve based methodology for the RPV (Reference 68).

A PBNP Unit 2 specific RT_{PTS} calculation was performed using the Master Curve methodology of BAW-2308, Revision 1, and is documented in Framatome ANP Calculation 32-5019743-01, "PBNP Unit 2 Power Uprate PTS Evaluation 53 EFPY," Revision 1, 08/19/2003 (Reference 74). Application of the Master Curve methodology for the Unit 2 RPV limiting girth weld demonstrates acceptable PTS values at EOEL.

Therefore, detailed plant-specific analyses can be submitted to demonstrate acceptable RT_{PTS} below the screening limit by application of new analysis techniques.

Aggressive Flux Reduction Program Option

Flux reduction strategies that required radical assembly design modifications were initially rejected during the original flux reduction program evaluations based on initial cost, increased fuel cycle costs, reduced core thermal design margins, increased possibility of fuel assembly damage, and an increased amount of material that needed to be disposed of as high-level nuclear waste.

One option that has the potential to achieve flux reduction factors approaching a factor of ten is the use of part length shielded fuel assemblies (PLSA) in the twelve peripheral fuel assembly locations on the core flats, i.e., along the core major axes. These are the locations where the Hafnium power suppression assemblies are currently located. This approach has been successfully implemented at the H. B. Robinson Plant at the onset of Cycle 10 in 1984.

Although the H. B. Robinson plant is a Westinghouse designed 3-loop reactor, the core configuration along the major axes is similar to the 2-loop configuration characteristic of Point Beach Unit 2. Both designs include twelve assemblies along the core flats with the main difference being the size of the individual assemblies. Further, for both reactor designs, the maximum pressure vessel fluence occurs along the major axes. Therefore, based on judgment, shielding effects similar to those observed at H. B. Robinson may be achievable at Point Beach. It should be noted that detailed feasibility studies of implementing this option at PBNP have not been performed.

In the design of the PLSA assemblies, the fuel pellets in a portion of the fuel assembly are replaced by stainless steel. This effectively removes the peripheral neutron source opposite the location of the limiting circumferential weld and, in addition, provides some shielding to prevent neutrons from the core interior from reaching the vessel.

In designing PLSA assemblies for potential application at Point Beach Unit 2, the specific axial location of the intermediate shell to lower shell circumferential weld must be taken into account. This location along with the desired degree of flux reduction will dictate the axial elevation of the center of the stainless steel pellet array, as well as, the total height of the steel region. Once a conceptual design of the PLSA assemblies is available, potential impacts on design, operation, and cost can be assessed.

One potentially significant advantage of a flux reduction plan based on the insertion of the PLSA assemblies is that the large flux reduction achieved by this approach allows the actual implementation to be scheduled well into the future. Assuming 1678 MWt, without Hafnium, and a flux reduction factor of 8, it appears feasible that Unit 2 could meet the current PTS acceptance criteria at EOEL if PLSAs were incorporated at EOL.

Pursuit of PLSA assemblies at this time is not considered reasonable or practical in view of cost, loss of core design margins, schedule requirement, and the capability of providing acceptable PTS results with alternate analysis techniques.

If acceptable PTS results cannot be provided prior to EOL with alternate analysis techniques, the PBNP flux reduction program will evaluate the feasibility and practicality of pursuing additional aggressive flux reduction measures prior to EOL, such as the insertion of part length shielded fuel assemblies.

Safety Analysis Option

10 CFR 50.61(b) (4) states "For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent RT_{PTS} from exceeding the PTS screening criterion using the EOL fluence, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least three years before RT_{PTS} is projected to exceed the PTS screening criterion."

Preliminary Plant Life Extension studies were performed to determine the viability of technical acceptance of the PBNP reactor vessels throughout a 20-year license renewal term. Westinghouse performed a comprehensive scoping risk assessment for the PBNP Reactor Vessels that is documented in WCAP-11676, "Scoping Risk Assessment for the PBNP Units 1 and 2 Reactor Vessel Life Extension Study", 1987. The Westinghouse report evaluated PBNP for PTS based on NRC Regulatory Guide 1.154. The analysis was performed without crediting any flux reduction measures. The report concluded that both PBNP Units 1 and 2 will be well within (by two orders of magnitude) the acceptance criteria of R.G. 1.154 throughout the 20-year license renewal term without any flux reduction measures being implemented. The fluence values used in the Westinghouse analysis conservatively envelope the current fluence projections.

Therefore, based on the results of the preliminary analysis, it is believed that a safety analysis in accordance with 10 CFR 50.61(b) (4) can be prepared which demonstrates acceptable risk from PTS for the license renewal term.

Annealing Option

10 CFR 50.61(b)(7) states "If the limiting RT_{PTS} value of the plant is projected to exceed the screening criteria in paragraph (b)(2), or the criteria in paragraphs (b)(3) through (b)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of § 50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the vessel beltline materials satisfy the requirements of paragraphs (b)(2) through (b)(6) of this section, with RT_{PTS} accounting for the effects of annealing and subsequent irradiation."

Thermal annealing of RPVs is a proven technology for recovering vessel material properties that have been degraded due to long-term exposure to neutron irradiation. Previous anneals have been performed principally in Eastern Europe on RPVs designed by the former Soviet Union.

Although there is no experience with annealing an operating commercial U.S. RPV, the Marble Hill Annealing Demonstration Project demonstrated that an anneal of a commercially sized U.S. RPV is technically feasible utilizing existing equipment and procedures.

Thus, annealing the PBNP Unit 2 RPV is a potential method to manage aging degradation associated with the loss of fracture toughness.

Conclusion

Maintenance of current power level, or continued operation with Hafnium suppression inserts in service is not necessary at this time since either the pursuit of an alternate fracture toughness evaluation methodology, pursuit of an aggressive flux reduction program in the future, pursuit of risk analysis, or pursuit of RPV annealing will provide technically acceptable methods of achieving EOEL with the PBNP RPVs.

The Reactor Vessel Surveillance Program will provide reasonable assurance that the Unit 2 RPV intermediate-to-lower shell girth weld PTS issue will be adequately managed for the period of extended operation in accordance with 10 CFR 50.61, per 10 CFR 54.21(c)(1)(iii).

4.2.2 Reactor Vessel Upper Shelf Energy

The requirements on reactor vessel Charpy upper-shelf energy are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any of the reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing. Limiting PBNP RPV weld materials fall below the

10 CFR 50, Appendix G, requirement of 50 ft-lb. Consequently, fracture mechanics evaluations were performed to demonstrate acceptable equivalent margins of safety against fracture.

The B&W Owners Group (B&WOG) performed equivalent margins analysis for B&W RPVs. The PBNP RPVs are included within the scope of the analyses. These analyses were performed assuming an original licensed power level, a low-low loading pattern with Hafnium, and EOL conditions. The analyses demonstrated acceptable equivalent margins of safety against fracture. The analyses are summarized in B&W Owners Reactor Vessel Working Group reports BAW-2178PA (Reference 63), "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," and BAW-2192PA (Reference 64), "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads," both dated April 1994. The NRC staff reviewed and approved both of these reports for referencing in licensing applications in separate safety evaluations on March 29, 1994 (Reference 65 and Reference 66).

Additional equivalent margins analyses have been performed for the PBNP RPVs to address the uprated power condition of 1678 MWt, without Hafnium power suppression absorber rods installed, and at EOEL conditions. The 2004 fluence projections were used to define EOEL vessel fluences. These analyses used the same methodologies described in the above references. The analyses, performed by Framatome ANP, are summarized in BAW-2467P, Revision 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years", October 2004 (Reference 75).

The revised analysis addresses ASME Levels A, B, C, and D Service Loadings. PBNP specific transient information was reviewed and incorporated into the plant specific analyses. There are no Level C service load transients specified for PBNP. For conservatism, three Level D transients were evaluated. These included the Reactor Coolant Line Break (LOCA), the FSAR Steam Line Break, and the RPV Equipment Specification Steam Line Break transients. The LOCA transient is the most limiting Level D transient.

For Levels A and B Service Loadings, the low upper-shelf toughness analysis is performed according to the acceptance criteria and evaluation procedures contained in Appendix K to Section XI of the ASME Code. The evaluation also utilizes the acceptance criteria and evaluation procedures prescribed in Appendix K for Levels C and D Service Loadings. Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of Framatome-ANP's PCRIT computer code to determine stress intensity factors for a worst case pressurized thermal shock transient.

The analysis shows that the ASME Code, Section XI, Appendix K acceptance criteria have been satisfied for Levels A, B, C, and D Service Loadings.

The limiting weld for the Upper Shelf Energy analysis is the SA-847 axial weld of the Point Beach Unit 1 RPV.

The analysis for Levels A and B service loadings shows that with factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J_1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1} / J_1$ = 1.87 which is significantly greater than the required value of 1.0. The analysis for Levels A and B service loadings also shows that with a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable.

The EOEL lower bounding J-R values and acceptance ratios for Levels A and B Service Loadings are summarized in Table 4.2.2-1.

Table 4.2.2-1
EOEL Lower Bounding J-R Values and Acceptance Ratios
Levels A and B Service Loadings

Unit	Weld Number	Weld Orientation	Lower Bounding	Acceptance Criterion 1		Acceptance Criterion 2	
			$J_{0.1}$ @1/4T (lb/in)	J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
Unit 1	SA-1101	Circ.	609	98	6.21	113	5.39
	SA-847	Long.	619	331	1.87	388	1.60
Unit 2	SA-1484	Circ.	579	104	5.57	119	4.87

The Unit 2 RPV intermediate-to-lower shell circumferential weld SA-1484 contains the minimum lower bounding J-R value at EOEL of 579 lb/in. The controlling weld is the Unit 1 RPV longitudinal weld SA-847. The minimum ratio of material J-R to applied J for acceptance criterion 1 and 2 at EOEL is 1.87 and 1.60 respectively. Since the values of the J-R ratios are greater than one, the acceptance criteria for the equivalent margins analysis have been met.

The analysis for Levels C and D service loadings shows that with a factor of safety of 1.0 on loading, flaw extensions are ductile and stable. The analysis for Levels C and D service loadings also shows that the flaw remains stable at much less than 75% of the vessel wall thickness. It has also been shown that the remaining ligament is sufficient to preclude tensile instability by a large margin.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.3 Reactor Vessel Pressure/Temperature Limits

Atomic Energy Commission (AEC) General Design Criterion (GDC) 14 of 10 CFR 50, Appendix A, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage or rapid failure and of gross rupture. Likewise, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed with sufficient margin to reasonably assure that when stressed by operation, maintenance, and testing conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires an appropriate materials surveillance program for assessing the structural integrity of the reactor vessel's beltline region.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, RT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The NRC has published a method for predicting irradiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

New heatup and cooldown pressure temperature (PT) limit curves have been developed for normal operation of PBNP Units 1 and 2 reactor pressure vessels through EOEL. The PT curves were generated based on the 2002 fluence projections assuming power uprating (1678 MWt) and the removal of the Hafnium absorber rods. Use of the 2002 fluence projections is conservative since these projections assumed power uprate occurred in 2002.

The heatup and cooldown curves were generated using the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 67) with the exception of the following: 1) The fluence values used in this report are calculated fluence values (i.e. comply with Reg. Guide 1.190), not the best estimate fluence values; 2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588); 3) The 1996 Version of Appendix G to Section XI will be used rather than the 1989 version; and 4) PT Curves were generated with the most limiting circumferential weld ART value in conjunction with Code Case N-588. These curves are bounded by the curves using the standard "axial" flaw methodology from ASME Code 1996 App. G with the ART from the intermediate or lower shell axial welds depending of the flaw location, 1/4T versus 3/4T.

The new Point Beach Unit 1 and 2 heatup and cooldown pressure-temperature limit curves were generated using adjusted reference temperature (ART) values that bound both units. The highest ART values from the two units were from the Unit 1 and Unit 2 intermediate to lower shell girth welds, however the limiting materials are actually the intermediate and lower shell axial welds from Unit 1, depending on the vessel thickness (1/4 T or 3/4 T location). The axial welds become limiting over the girth weld through use of "circ-flaw" methodology from ASME Code Case N-588. This methodology is less restrictive than the standard "axial-flaw" methodology from the 1995 ASME Code, Section XI through the 1996 Addenda. In addition to the use of Code Case N-588, the PT curves also made use of ASME Code Case N-640, which allows the use of the K_{Ic} methodology. Both ASME Code Case N-588 and N-640 were joined together under ASME Code Case N-641.

The calculation of heat-up and cool-down curves requires ART values at the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) through wall locations corresponding to the peak fluence for the girth (circumferential) weld. The attenuation of fluence through the wall of the RPV was determined using the method in Regulatory Guide 1.99, Rev. 2.

Contained in Table 4.2.3-1 is a summary of the limiting ARTs used in the generation of the PBNP Units 1 and 2 PT limit curves.

Table 4.2.3-1
Summary of the Limiting ART Values Used in the Generation of the PBNP Units 1 and 2 Heatup/Cooldown Curves

EFPY	Limiting "Circ-Flaw" ART (i)		Limiting "Axial-Flaw" ART (°F)	
	¼T(°F)	¾T (°F)	¼T(°F)	¾T (°F)
PBNP Unit 1 (ii)				
53	286	254	243	224
PBNP Unit 2 (iii)				
53	301	267	152	140

i. PBNP Units 1 and 2 Limiting Circ. Flaw ART comes from the Intermediate to lower shell circumferential welds (Heat #'s 71249 and 72442, respectively)
ii.The "Axial-Flaw" ARTs for PBNP Unit 1 are from the lower shell axial welds (1/4T) and the intermediate shell axial welds (3/4T)
iii.The "Axial-Flaw" ARTs for PBNP Unit 2 are from the intermediate shell forging 123V500.

Limiting heatup curves were generated using heatup rates of 60 and 100°F/hr for 53 EFPY (EOEL). These curves were generated using a combination of the 1996 ASME Code Section XI, Appendix G with the limiting ART values from the Unit 1 intermediate and lower shell longitudinal welds and the ASME Code Case N-641. These heatup curves bound those generated using the "Circ-flaw" methodology portion of ASME Code Case N-641 with the limiting circ-weld ART values from the Unit 1 or 2 intermediate to lower shell girth weld.

Limiting cooldown curves were generated using cooldown rates of 0, 20, 40, 60 and 100°F/hr for 53 EFPY (EOEL). Again, these curves were generated using a combination of the1996 ASME Code Section XI, Appendix G with the limiting ART values from the Unit 1 intermediate and lower shell longitudinal welds and the ASME Code Case N-641. These cooldown curves bound those generated using the "Circ-flaw" methodology portion of ASME Code Case N-641 with the limiting circ-weld ART values from the Unit 1 or 2 intermediate to lower shell girth weld.

The PT curves include a hydrostatic leak test limit curve from 2485 psig to 2000 psig, along with the pressure-temperature limits for the vessel flange region per the requirements of 10 CFR 50, Appendix G. A copy of these PT limit curves was provided to the NRC in Reference 70.

In addition, maximum allowable low-temperature, overpressure protection system (LTOPS) power-operated relief valve (PORV) lift setpoints have been developed for 53 EFPY (EOEL), based on the P-T limits applicable to the period of operation.

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

Appendix A Revisions

The proposed FSAR Aging Management Program discussion as provided in Appendix A of the License Renewal Application is as follows.

15.2.18 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessels. Monitoring methods will be in accordance with 10 CFR 50, Appendix H. This program includes (a) capsule insertion, withdrawal and materials testing/evaluation, (including upper shelf energy and RT_{NDT} determinations), (b) fluence and uncertainty calculations, (c) monitoring of Effective Full Power Years (EFPY), (d) development of pressure-temperature limitations, (e) determination of low-temperature overpressure protection (LTOP) set points, and (f) implementation of a flux reduction program, and other options as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 intermediate-to-lower shell girth weld. The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61.

Section 15.4 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement. These discussions are numbered and inserted into the FSAR sections where these subjects are covered.

15.4.1 Reactor Vessel Irradiation Embrittlement

PBNP Units 1 and 2 reactor vessels are described in Chapters 3.0 and 4.0. Time-limited aging analyses (TLAAs) applicable to the reactor vessels are:

- Pressurized thermal shock
- Upper-shelf energy
- Pressure-temperature limits

The Reactor Vessel Surveillance Program manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time-dependent parameters used in the aging analyses for pressurized thermal shock, upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

Reactor Vessel Pressurized Thermal Shock

The requirements in 10 CFR 50.61 provide rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of the maximum nil ductility reference temperature (RT_{PTS}) whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility.

The calculated RT_{PTS} values at the end of life extension for the PBNP Units 1 and 2 reactor vessels are less than the 10 CFR 50.61(b)(2) screening criteria of 270°F for intermediate and lower shells and 300°F for the circumferential welds, with the exception of the Unit 2 RPV intermediate-to-lower shell circumferential weld.

The EOEL fluence yields an RT_{PTS} value of 316°F when using Charpy based methods for the limiting weld of the Unit 2 RPV. The screening criteria established in 10 CFR 50.61 (300°F) will be exceeded for the limiting Unit 2 intermediate to lower shell girth weld at a neutron fluence of 3.31×10^{19} n/cm^2. The 2004 fluence projections indicate that the limiting weld will experience this fluence at 38.1 EFPYs. Assuming a long-term capacity factor of 95 %, this fluence would be achieved late in 2017.

The PBNP Reactor Vessel Surveillance Program includes a flux reduction program to manage the Unit 2 RPV intermediate-to-lower shell girth weld PTS issue for the period of extended operation in accordance with 10 CFR 50.61(b)(3).

The current PBNP flux reduction actions will not prevent the intermediate-to-lower shell girth weld from exceeding the PTS screening criteria at EOEL.

The PBNP Reactor Vessel Surveillance Program will include other options to manage Reactor Vessel Integrity per 10 CFR 50.61. These options will include consideration of an alternate fracture toughness evaluation methodology, pursuit of an aggressive flux reduction program in the future, pursuit of risk analysis, or pursuit of RPV annealing. Each of these options can provide technically acceptable methods of achieving EOEL with the PBNP RPVs.

The Reactor Vessel Surveillance Program will provide reasonable assurance that the Unit 2 RPV intermediate-to-lower shell girth weld PTS issue will be adequately managed for the period of extended operation in accordance with 10 CFR 50.61, per 10 CFR 54.21(c)(1)(iii).

Use of the Master Curve methodology, extrapolated to EOEL fluence, shows that the RPV limiting weld metal meets PTS screening criteria out to EOEL and beyond. These projections will be confirmed by additional testing of weld heat 72442 from the B&W Owners Group MIRVP prior to reaching the EOL fluence at PBNP Unit 2. A supplemental surveillance program will be designed and implemented at PBNP Unit 2 that includes the limiting weld metal for future evaluation using the Master Curve methodology. The testing of this supplemental capsule at a fluence corresponding to EOEL will confirm the toughness condition for the PBNP Unit 2 RPV weld at about 38 EFPY, which is well before EOEL is reached.

The analysis associated with pressurized thermal shock has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Reactor Vessel Upper-Shelf Energy

The requirements on reactor vessel Charpy upper-shelf energy are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any reactor vessel material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

A fracture mechanics evaluation was performed in accordance with Appendix K of ASME Section XI to demonstrate continued acceptable equivalent margins of safety against fracture through the end of life extension.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Reactor Vessel Pressure/Temperature Limits

The requirements in 10 CFR 50, Appendix G, ensure that heatup and cooldown of the reactor pressure vessel are accomplished within established pressure-temperature limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced.

Operation of the Reactor Coolant System is also limited by the net positive suction head curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the reactor coolant pumps net positive suction head curves.

To address the period of extended operation, the end of license extension projected fluences, and the RPV material properties were used to determine the limiting materials, and calculate pressure-temperature limits for heatup and cooldown. The new Point Beach Unit 1 and 2 heatup and cooldown pressure-temperature limit curves were generated using adjusted reference temperature (ART) values that bound both units. The highest ART values from the two units were from the Unit 1 and Unit 2 intermediate-to-lower shell girth welds, however the limiting materials are actually the intermediate and lower shell axial welds from Unit 1, depending on the vessel thickness (1/4 T or 3/4 T location). The axial welds become limiting over the girth weld through use of "circ-flaw" methodology from ASME Code Case N-588. This methodology is less restrictive than the standard "axial-flaw" methodology from the 1995 ASME Code, Section XI through the 1996 Addenda. In addition to the use of Code Case N-588, the PT curves also made use of ASME Code Case N-640, which allows the use of the K_{Ic} methodology. Both ASME Code Case N-588 and N-640 were joined together under ASME Code Case N-641.

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

15.5 Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No existing TLAA related exemptions were identified.

Appendix B Revisions

The Reactor Vessel Surveillance Program as provided in Appendix B of the License Renewal Application is as follows.

B2.1.18 Reactor Vessel Surveillance Program

Program Description

The Reactor Vessel Surveillance Program manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessels. Monitoring methods will be in accordance with 10 CFR 50, Appendix H. This program includes (a) capsule insertion, withdrawal and materials testing/evaluation, (including upper shelf energy and RT_{NDT} determinations), (b) fluence and uncertainty calculations, (c) monitoring of Effective Full Power Years (EFPY), (d) development of pressure-temperature limitations, (e) determination of low-temperature overpressure protection (LTOP) set points, and (f) implementation of a flux reduction program, and other options as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 intermediate-to-lower shell girth weld. The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61.

The Reactor Vessel Surveillance Program consists of six major subprograms:

- Surveillance Capsule Insertion, Withdrawal, and Evaluation,
- Fluence and Uncertainty Calculations,
- Monitoring of Effective Full Power Years (EFPY),
- Development of Pressure-Temperature Limit Curves,
- Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints, and
- Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

NUREG-1801 Consistency

The Reactor Vessel Surveillance Program is an existing program that is consistent with, but includes exceptions to, NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance" (Reference 3). The Reactor Vessel Surveillance Program is also an existing program that consists of the appropriate ten elements described in Branch Technical Position RLSB-1, "Aging Management Review-Generic," which is included in Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

Exceptions to NUREG-1801

See the following element discussion for elaboration on the exceptions to the NUREG-1801 aging management program element assumptions:

- Acceptance Criteria

Enhancements

Enhancements to the Reactor Vessel Surveillance Program include changes to the FSAR and TRM to reflect the materials and withdrawal schedule of the new surveillance capsule and revisions to plant procedures to clarify organizational responsibilities, describe the plan/schedule for removal, testing and evaluation of surveillance capsules, and implement a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 intermediate-to-lower shell girth weld.

These enhancements are required to satisfy the NUREG-1801 aging management program requirements. Details of the enhancements are included in the appropriate element descriptions below.

Implementation of a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61 (b) for the Unit 2 RPV intermediate-to-lower shell girth weld will be completed within 1-year of receipt of the extended license. Implementation within this time frame will support submittal of any required safety analysis at least three years prior to the time that RT_{PTS} for Unit 2 is projected to exceed the screening criteria. The other enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The key elements, which are used in the Reactor Vessel Surveillance Program, are described below. The results of an evaluation of each key element against NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance," is provided below. An evaluation of each key element against the appropriate ten elements described in Branch Technical Position RLSB-1, "Aging Management Review-Generic," which is included in Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," was also conducted.

Elements of the first three listed subprograms (i.e. Surveillance Capsule Insertion, Withdrawal and Evaluation, Fluence and Uncertainty Calculations, and Monitoring of Effective Full Power Years) are addressed by the NUREG-1801 program. The subprograms for the Calculation and Monitoring of LTOP Setpoints, Development of Pressure-Temperature Limit Curves, and Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2 are not addressed by the NUREG-1801 program.

Scope of Program

The Reactor Vessel Surveillance Program consists of PBNP activities that manage the aging effects for components in the following systems and structures:

Reactor Vessel

The Reactor Vessel Surveillance Program only applies to the PBNP-1 and PBNP-2 reactor pressure vessels.

Surveillance Capsule Insertion, Withdrawal, and Evaluation

The program controls the development of surveillance capsule insertion and withdrawal schedules and capsule materials testing. Although the original surveillance capsules did not contain the most limiting material with respect to embrittlement, an additional surveillance capsule was installed in 2002 that contains the most limiting material. The surveillance program therefore meets the requirements of ASTM E 185-82.

The capsule installed in 2002 will be withdrawn during an outage at which it has accumulated a fluence equivalent to the 60-calendar year vessel fluence. Data from an integrated surveillance program that includes all PWRs with reactor vessels fabricated by B&W will also be used to predict embrittlement. Spare capsules remaining in both the PBNP-1 and PBNP-2 reactor vessels do not contain the most limiting materials and there are no current plans to withdraw these capsules.

The results of capsule materials testing, fluence analysis, and EFPY monitoring are used to predict the effects of neutron embrittlement through the end of extended life (EOEL). Prediction of the effects of radiation on reactor vessel beltline materials is in accordance with RG 1.99, Revision 2. Both the chemistry tables (RG 1.99, Revision 2, Position 1) and surveillance data (RG 1.99, Revision 2, Position 2) are used to project embrittlement. The limitations of RG 1.99, Revision 2, Position 1.3 are observed for material properties, temperature, material chemistry, and fluence.

The results of capsule tests, fluence analysis, and EFPY monitoring are also used to determine compliance with the PTS screening criteria of 10 CFR 50.61. A flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) will be implemented if the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criteria using the EOEL fluence.

Fluence and Uncertainty Calculations

Calculations are performed for the PBNP-1 and PBNP-2 reactor vessels in accordance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The results are used as an input to embrittlement predictions.

Monitoring of Effective Full Power Years (EFPY)

EFPY monitoring is accomplished using operations data for the PBNP-1 and PBNP-2 reactors. The results are used to project the fluence corresponding to specific values of EFPY.

Development of Pressure-Temperature Limit Curves

The Reactor Vessel Surveillance Program controls the development of pressure and temperature limit curves in accordance with 10 CFR 50, Appendix G requirements. The methods of ASME Section XI, Appendix G are used to determine pressure and temperature limits. The fracture toughness used in calculating P-T limits is determined as a function of the difference in temperature from RT_{NDT}. RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" is used to determine RT_{NDT}. ASME Code Case N-641 allows the use of the K_{IC} curve, an alternate fracture toughness curve to the K_{IR} curve, which is a modification to the acceptance criteria of ASME Section XI, Appendix G.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

The Reactor Vessel Surveillance Program requires the calculation of LTOP set points for the PBNP-1 and PBNP-2 reactor coolant systems. These set points ensure that an LTOP event will not increase the probability of brittle fracture of the reactor vessels. LTOP set points include the maximum pressure allowed before the LTOP system actuates to relieve the pressure, and the temperature below which the LTOP system must be effective. These pressures and temperatures are determined using the method of ASME Section XI, Appendix G or using an alternative method provided by ASME Code Case N-641.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

Because the RT_{PTS} value of the Unit 2 RPV intermediate-to-lower shell girth weld is projected to exceed the PTS screening criteria prior to the EOEL, a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) will be implemented on Unit 2. Ex-vessel neutron dosimetry sets, installed in the reactor cavity annulus, may be used to verify analytical flux reduction predications of the flux reduction program.

This element is consistent with the NUREG-1801 aging management program.

Preventive Actions

This surveillance program determines neutron embrittlement for upper-shelf energy and pressure-temperature limits for 60 years in accordance with the RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Surveillance Capsule Insertion, Withdrawal, and Evaluation

Surveillance Capsule Insertion, Withdrawal, and Evaluation do not constitute preventive actions.

Fluence and Uncertainty Calculations

Fluence and uncertainty calculations do not constitute preventive actions.

Monitoring of Effective Full Power Years (EFPY)

EFPY monitoring is a monitoring activity, not a preventive action.

Development of Pressure-Temperature Limit Curves

The development of, and operation within, P-T limit curves minimizes the probability of brittle fracture of the reactor vessel during normal operation.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

The LTOP system with the actuation setpoints and operational restrictions established by the LTOP analysis, minimizes the probability of an LTOP event, and therefore, helps to minimize the probability of reactor vessel brittle fracture.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

Because the RT_{PTS} value of the Unit 2 RPV intermediate-to-lower shell girth weld is projected to exceed the PTS screening criteria prior to the EOEL, a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) will be implemented. The flux reduction program and other options, as necessary, allowed by 10 CFR 50.61 will ensure that the probability of brittle fracture of the reactor vessels during a PTS event is acceptably low.

This element is consistent with the NUREG-1801 aging management program.

Parameters Monitored or Inspected

Surveillance Capsule Insertion, Withdrawal, and Evaluation

The program monitors the effects of neutron irradiation on the PBNP-1 and PBNP-2 reactor vessel beltline materials. Fracture toughness of beltline materials is indirectly monitored through measurement of the impact energy of Charpy V-Notch (CV) specimens, made from representative materials from the PBNP reactor vessels beltline regions. CV test results from capsules irradiated in other PWRs participating in an integrated surveillance program are also used to aid in trending the change in material properties of the PBNP reactor vessels. Fracture toughness specimens to be irradiated in the PBNP-2 vessel and in the Master Integrated Reactor Vessel Surveillance Program (MIRVSP) will be withdrawn and tested. The surveillance capsules also contain neutron dosimetry that monitors the amount of neutron fluence received by the test specimens.

Fluence and Uncertainty Calculations

This subprogram does not monitor, inspect, or test any parameters. Neutron fluence measurements acquired under the surveillance capsule insertion, withdrawal and testing subprogram are used to validate analytical models that determine the fluence received by the reactor vessel.

Monitoring of Effective Full Power Years (EFPY)

Effective Full Power Years (EFPY) are monitored and used to predict the fluence that the vessel will accumulate at some future time, which is then used to predict change in RT_{NDT} and upper shelf energy (USE).

Development of Pressure-Temperature Limit Curves

No parameters are monitored or inspected under this subprogram.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

LTOP system relief valve operation is monitored to determine whether an LTOP event could have occurred had the LTOP system been inoperable. Operation within the P-T limits is also monitored.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

Ex-vessel neutron dosimetry sets, installed in the reactor cavity annulus, may be used to verify analytical flux reduction predications of a flux reduction program.

This element is consistent with the NUREG-1801 aging management program.

Detection of Aging Effects

Surveillance Capsule Insertion, Withdrawal, and Evaluation

Aging effects are detected through testing of surveillance materials. CV tests are performed to determine the decrease in USE and increase in transition temperature RT_{NDT}, for materials that closely match reactor vessel beltline materials.

Fluence and Uncertainty Calculations

This subprogram does not detect aging effects.

Monitoring of Effective Full Power Years (EFPY)

This subprogram does not detect aging effects.

Development of Pressure-Temperature Limit Curves

This subprogram does not detect aging effects.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

This subprogram does not detect aging effects.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

This subprogram does not detect aging effects.

Enhancements will be made to revise the surveillance capsule withdrawal schedule contained in the PBNP FSAR and TRM to reflect the planned withdrawal of the new surveillance capsule that was installed in PBNP-2 during the 2002 refueling outage. A description of the materials included in this capsule, including fracture toughness specimens, must also be added to the PBNP FSAR. In addition, plant procedures will be modified as follows:

- Add a requirement that the reactor vessel engineer shall ensure that all withdrawn surveillance capsules not discarded as of August 31, 2000, are placed in storage for the purposes of future reconstitution and use, if necessary.

- Add a requirement that the reactor vessel engineer shall ensure that the number of EFPY accrued by PBNP-1 and PBNP-2 is updated by January 1 of each year.

- Add a requirement that the reactor vessel engineer shall ensure that the fluence and uncertainty calculations for PBNP-1 and PBNP-2 are updated periodically. The reactor vessel engineer should trend the rate of fluence accumulation versus EFPY. Based on the updated projection of fluence versus EFPY, the reactor vessel engineer shall review the number of EFPY associated with the expiration of the current P-T limits to determine if this projected amount of EFPY remains valid.

- Add a requirement that a determination of the number of EFPY accumulated by January 1 of the current year shall be performed and documented annually.

- Provide a description of the existing equivalent margins analyses for low USE that was performed for PBNP Unit-1 and PBNP Unit-2 with projected fluence values for EOEL (53 EFPY) assuming power uprate (1678 MWt) conditions without Hafnium power suppression assemblies installed.

- Specify that the methods of RG 1.99, Revision 2, are used to demonstrate compliance with the fracture toughness requirements of 10 CFR 50, Appendix G.

- Add a description of the methodology of fluence and uncertainty calculations.

- Describe the implementation of a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 intermediate-to-lower shell girth weld.

- Add a requirement to install neutron dosimetry if the last surveillance capsule in PBNP -2 is withdrawn prior to the 55th year of operation.

- Add a description of the plan/schedule for removal, testing and evaluation of surveillance capsules.

This element is consistent with the NUREG-1801 aging management program.

Monitoring and Trending

Surveillance Capsule Insertion, Withdrawal, and Evaluation

Monitoring of reactor vessel beltline fracture toughness is accomplished through testing of surveillance specimens from surveillance capsules that are periodically withdrawn from the vessels. Trending is accomplished through the RG 1.99, Revision 2 methods for projection of RT_{NDT} and USE. Projection of the increase in RT_{NDT} and the decrease in USE provides early indication if the fracture toughness properties of the PBNP reactor vessel beltline materials will fail to meet regulatory requirements. The RT_{PTS} projection is compared to the PTS screening criteria of 270°F for plates, forgings, and axial welds, and 300°F for circumferential welds specified in 10 CFR 50.61. USE projections are compared against the requirement to maintain 50 ft-lbs or greater given by 10 CFR 50, Appendix G.

Fluence and Uncertainty Calculations

Fluence measurements from capsules are trended to verify that actual fluence is adequately represented by fluence models and to project fluence for future dates. A surveillance capsule containing neutron dosimetry or some form of neutron dosimetry, will remain installed in the reactor vessels until at least the 55th year of operation.

Monitoring of Effective Full Power Years (EFPY)

EFPY are monitored and trended to allow the EFPY for particular calendar dates, such as the end of the current and extended license periods, to be projected, and to establish deadlines for revising P-T curves that are valid only to a particular number of EFPY. These projections will be extended to a number of EFPY corresponding to the end-of-extended-life (EOEL).

Development of Pressure-Temperature Limit Curves

This subprogram does not perform trending, but relies on the trending of changes in material properties, fluence, and EFPY to set limits on P-T curve validity.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

PBNP monitors and trends actuation of relief valves relied on for LTOP protection, to determine if the actuation is a reportable event. This subprogram relies on the trending of changes in material properties, fluence, and EFPY, to determine the inputs to calculations of LTOP set points.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

Ex-vessel neutron dosimetry sets, installed in the reactor cavity annulus, may be used to verify analytical flux reduction predications of a flux reduction program.

This element is consistent with the NUREG-1801 aging management program.

Acceptance Criteria

Surveillance Capsule Insertion, Withdrawal, and Evaluation

The upper shelf energy of the most limiting material in the reactor vessel beltline must remain above 50 ft-lbs until the end-of-extended-life, using the methods of RG 1.99, Revision 2 with the PBNP specific and integrated surveillance program data as inputs or equivalent margin demonstrated. The RT_{PTS} of the most limiting material in the reactor vessel beltline must not exceed the PTS screening criteria specified by 10 CFR 50.61 (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds), unless it can be demonstrated by alternate means, as allowed by 10 CFR 50.61, that the probability of brittle fracture of the reactor vessel in a PTS event is acceptably low.

Fluence and Uncertainty Calculations

These calculations do not have specific acceptance criteria.

Monitoring of Effective Full Power Years (EFPY)

This activity does not have specific acceptance criteria. EFPY monitoring does affect the validity of the pressure-temperature limit curves, which are linked to a specific range of EFPY.

Development of Pressure-Temperature Limit Curves

The acceptance criteria for P-T curves is that the flaw stability criteria of ASME Section XI, Appendix G, are met for all normal operating conditions as required by 10 CFR 50, Appendix G. The acceptance criteria of Appendix G may be modified through application of ASME Code Case N-641, which allows the use of the K_{IC} curve, an alternate fracture toughness curve to the K_{IR} curve. Pressure-temperature curves are acceptable only through a specific value of EFPY that is based on a fluence projection for that number of EFPY.

Calculation and Monitoring of Low Temperature Overpressure Protection (LTOP) Setpoints

LTOP set points are acceptable only through a specific value of EFPY that is based on a fluence projection for that number of EFPY.

Implementation of a Flux Reduction Program and 10 CFR 50.61(b) Options for Unit 2

If no reasonably practical flux reduction program can be shown to prevent RT_{PTS} from exceeding the PTS screening criteria prior to EOEL, other options allowed by 10 CFR 50.61(b) will be evaluated and implemented.

This element includes exceptions to the NUREG-1801 aging management program. NUREG-1801 does not provide for use of ASME Code Case N-641 fracture toughness curves when calculating P-T limit curves.

PBNP meets the intent of this NUREG-1801 aging management program.

Corrective Actions

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as committed in Section 1.4 of the PBNP Final Safety Analysis Report (FSAR).

This element is consistent with the NUREG-1801 aging management program.

Confirmation Process

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B and ANSI N18.7-1976, as committed in Section 1.4 of the PBNP FSAR.

This element is consistent with the NUREG-1801 aging management program.

Administrative Controls

The Reactor Vessel Surveillance Program is implemented through various plant documents. These implementing documents are subject to administrative controls, including a formal review and approval process, in accordance with the requirements of 10 CFR 50, Appendix B and ANSI N18.7-1976, as committed in Section 1.4 of the PBNP FSAR.

This element is consistent with the NUREG-1801 aging management program.

Operating Experience

PBNP-1 and PBNP-2 have generally operated successfully within their licensed P-T limits. New P-T curves are developed and issued, as required. An event involving the actuation of the LTOP system relief valves at PBNP-1 occurred on October 23, 1997. The event was evaluated and the conclusion was that an over pressurization event would have occurred if the LTOP system had been inoperable. A report to the NRC was therefore required. However, the LTOP system functioned correctly, preventing the over pressurization. The calculation also took no credit for manual operator action that may have prevented the over pressurization.

PBNP-1 will continue to meet the requirements of 10 CFR 50, Appendix G and 10 CFR 50.61 through the end of extended life. RT_{PTS} for the intermediate-to-lower shell girth weld in the PBNP-2 vessel is predicted to exceed the PTS screening criteria prior to the end of license extension and will be addressed through a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) as part of the Reactor Vessel Surveillance Program.

The program has been modified to incorporate data from the B&W integrated surveillance program. A replacement surveillance capsule containing materials closely matching the limiting materials for both PBNP-1 and PBNP-2 has been installed in the PBNP-2 reactor vessel during the 2002 refueling outage. The selection of materials for this capsule reflects the evolution in the understanding of the variables that control embrittlement of reactor pressure vessel steels, which resulted in a reassessment of the identity of the limiting materials in the PBNP-1 and PBNP-2 vessels.

Industry operating experience related to the Reactor Vessel Surveillance Program includes GL 92-01, Revision 1, "Reactor Vessel Structural Integrity," and Supplement 1 to GL 92-01, Revision 1, "Reactor Vessel Structural Integrity." PBNP's response to these documents has been incorporated into the Reactor Vessel Surveillance Program.

A review of NRC Inspection Reports, QA Audit/Surveillance Reports, and Self-Assessments since 1999 revealed no issues or findings that could impact the effectiveness of the Reactor Vessel Surveillance Program. The Second Quarter 2000 Engineering Audit assessed the reactor vessel integrity program, as controlled by plant procedures. The audit examined several activities including a calculation of the date at which the neutron fluence would exceed the limits of the current P-T curves, the progress of a submittal to the NRC of revised P-T curves, and a calculation of the LTOP applicability date. These activities were found to have been completed satisfactorily. The auditors found that corrective actions related to previous Condition Reports had been completed and there were no new Condition Reports. The auditors therefore judged the program to be effective. As additional operating experience is obtained, lessons learned may be used to adjust this program.

This element is consistent with the NUREG-1801 aging management program.

Conclusion

The Reactor Vessel Surveillance Program provides reasonable assurance that the aging effects will be managed consistent with the current licensing basis for the period of extended operation. The Reactor Vessel Surveillance Program complies with the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50, Appendices G and H. The combination of the PBNP original surveillance program and the B&W Integrated Surveillance Program has been used to demonstrate that the reference temperature for the PBNP-1 reactor vessel limiting beltline materials will not exceed the PTS screening criteria of 10 CFR 50.61 prior to the end-of-extended life. Because the PBNP-2 reactor vessel intermediate-to-lower shell girth weld reference temperature is predicted to exceed the PTS screening criteria prior to the end of license extension, a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) will be included for Unit 2 as part of the Reactor Vessel Surveillance Program.

The upper shelf energy (USE) for the limiting materials of the PBNP-1 and PBNP-2 reactor vessels is projected to fall below 50 ft-lbs by the end of the current license. PBNP has performed analyses that demonstrate equivalent margins against ductile fracture to those required by 10 CFR 50, Appendix G, through the end-of-extended-life.

To further refine the predictions of the material properties at the end-of-extended life (corresponding to 60 calendar years), an additional surveillance capsule containing materials that closely match the limiting materials in the reactor vessel beltline of both PBNP-1 and PBNP-2, has been installed in PBNP-2. This capsule will be withdrawn after it has received a fluence equivalent to the vessel fluence at 60 calendar years.

References

Reference 75: Framatome ANP, Inc., AREVA and Siemens Company Calculation, BAW-2467P, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years, October 2004.

ENCLOSURE 2

BAW–2467NP, Revision 1

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSEL OF POINT BEACH UNITS 1 AND 2 FOR EXTENDED LIFE THROUGH 53 EFFECTIVE FULL POWER YEARS

OCTOBER 2004

Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years

AREVA Document No. 77-2467NP-01
(See Section 11 for document signatures.)

Prepared for

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EXECUTIVE SUMMARY

Nuclear Management Company is considering plant life extension, power uprate to 1678 MWt and removal of hafnium power suppression assemblies from the core for Point Beach Units 1 and 2. As a result of these changes, operating conditions including vessel temperatures and projected fluence values at 53 effective full power years (EFPY) of plant operation have changed. It must be ensured that these changes do not affect the plant adversely from a regulatory compliance point of view. One of the compliance issues is Appendix G to 10 CFR Part 50 where low upper-shelf toughness is addressed. An equivalent margins assessment has to be made for material toughness when the upper-shelf Charpy energy level falls below 50 ft-lb. This report addresses this particular compliance issue regarding low upper-shelf toughness only.

The Charpy upper-shelf value of reactor vessel beltline weld materials at Point Beach Units 1 and 2 may be less than 50 ft lb at 53 EFPY. In order to demonstrate that sufficient margins of safety against fracture remain to satisfy the requirements of Appendix G to 10 CFR Part 50, a low upper-shelf toughness fracture mechanics analysis has been performed. The limiting welds in the beltline region have been evaluated for ASME Levels A, B, C, and D Service Loadings based on the evaluation acceptance criteria of the ASME Code, Section XI, Appendix K.

The analysis presented in this report demonstrates that the limiting reactor vessel beltline weld at Point Beach Units 1 and 2 satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 53 EFPY.

RECORD OF REVISIONS

Revision	Affected Pages	Description	Date
0	All	Original release	July 2004
1	All	Updated fluence values used for Evaluation Condition 1	October 2004

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1.0 Introduction

Nuclear Management Company is considering plant life extension, power uprate to 1678 MWt and removal of hafnium power suppression assemblies from the core for Point Beach Units 1 and 2. This document assesses the effect of these proposed changes on the upper-shelf fracture toughness of the reactor vessels. The B&W Owners Group (B&WOG) fracture toughness model was used in the low upper-shelf toughness fracture mechanics analyses of the reactor vessels of the B&WOG Reactor Vessel Working Group (RVWG) which includes the Point Beach Units 1 and 2 reactor vessels. The low upper-shelf toughness analysis for all reactor vessels of the B&WOG RVWG for Levels A & B Service Loadings was documented in BAW-2192PA [1]. An additional fracture mechanics analysis for Levels C & D Service Loadings was carried out for all these reactor vessels and documented in BAW-2178PA [2]. Both these reports have been accepted by the NRC. As a result of a subsequent power uprate, an additional low upper-shelf toughness analysis covering end-of-license and end-of-license renewal fluence values was performed for Point Beach Units 1 and 2 [3]. For the current planned changes, the effect on the reactor vessel materials upper-shelf toughness is assessed in this report.

Welds in the beltline region of all B&W Owners Group Reactor Vessel Working Group plants, including the Point Beach Units 1 and 2 vessels, have been analyzed [1, 2] for 32 effective full power years (EFPY) of operation to demonstrate that these low upper-shelf energy materials would continue to satisfy federal requirements for license renewal. In Reference 3, the Point Beach vessels were analyzed up to their forecasted end-of-license extension periods at a partially uprated power level of 1650MWt with hafnium power suppression assemblies, and both vessels were shown to be acceptable. The purpose of the present analysis is to perform a similar low upper-shelf toughness evaluation of the reactor vessel welds at the Point Beach plants for projected neutron fluences at 53 EFPY.

The present analysis addresses ASME Levels A, B, C, and D Service Loadings. For Levels A and B Service Loadings, the low upper-shelf toughness analysis is performed according to the acceptance criteria and evaluation procedures contained in Appendix K to Section XI of the ASME Code [4]. The evaluation also utilizes the acceptance criteria and evaluation procedures prescribed in Appendix K for Levels C and D Service Loadings. Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of Framatome ANP's PCRIT computer code to determine stress intensity factors for a worst case pressurized thermal shock transient.

Revision 1 of this document utilizes the updated fluence values calculated in 2004 for the uprated power condition of 1678 MWt without the hafnium power suppression assemblies installed. This input was provided by the Nuclear Management Company (NMC) and is included as Appendices A and B.

2.0 Changes In Operating Condition Parameters

As a result of the planned updates to the Point Beach Units 1 and 2, there are increases in the projected end of life fluences for both the units. There are also changes in the plants' operating temperatures. These inputs were provided by the Nuclear Management Company and included as Appendices A and B and summarized in this section.

The analysis for current licensed rated power conditions (1540 MWt) gives a maximum cold leg temperature of 544.5°F. As a result of the power uprate to 1678 MWt, the maximum cold leg temperature is reduced to 541.4°F. The projected reactor vessel fluence values at 53 EFPY are provided in Table 2-1. For this analysis, three cases, termed Evaluation Conditions, are studied – uprated power conditions without hafnium assemblies, current power conditions without hafnium assemblies, and current power conditions with hafnium assemblies. Fluence values for these three cases are reported only for the controlling welds identified through review of the results reported in References 1, 2 and 3. Locations of the reactor vessel welds for Point Beach Units 1 and 2 are illustrated in Figures 2-1 and 2-2 respectively [1].

Table 2-1 Evaluation Conditions

Plant	Weld Location [1]	Weld Number [1]	Cu (wt%) [5]	Ni (wt%) [5]	Fluence (n/cm^2) at 53 EFPY		
					EVALUATION CONDITION 1 Uprated Power Conditions Without Hafnium Assemblies Cold Leg Temp: 541.4°F	EVALUATION CONDITION 2 Current Power Conditions Without Hafnium Assemblies Cold Leg Temp: 544.5°F	EVALUATION CONDITION 3 Current Power Conditions With Hafnium Assemblies Cold Leg Temp: 544.5°F
PB-1	Lower Shell Long.	SA-847	0.23	0.52	3.25E+19	3.12E+19	2.67E+19
	Inter. Shell/Lower Shell Circ.	SA-1101	0.23	0.59	4.71E+19	4.52E+19	3.82E+19
PB-2	Inter. Shell/Lower Shell Circ.	SA-1484	0.26	0.60	4.85E+19	4.65E+19	3.79E+19

Figure 2-1 Reactor Vessel of Point Beach Unit 1 [1]

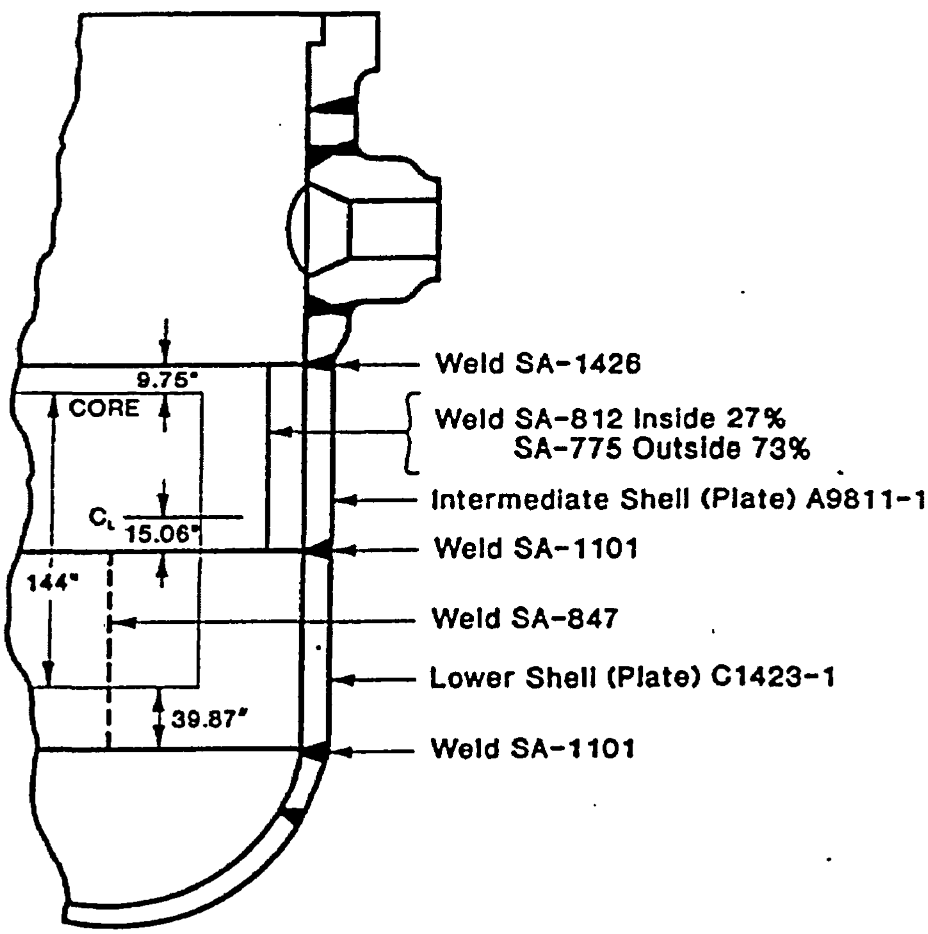

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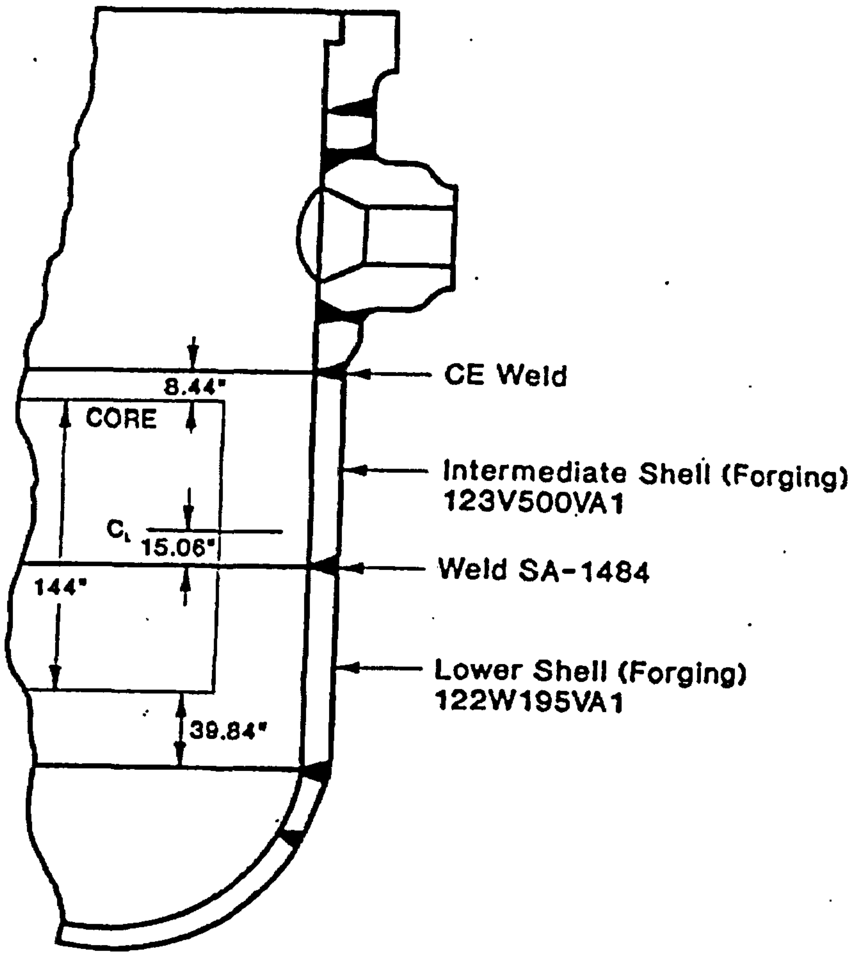
Figure 2-2 Reactor Vessel of Point Beach Unit 2 [1]



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3.0 Material Properties and Reactor Vessel Design Data

An upper-shelf fracture toughness material model is discussed below,. as well as mechanical properties for the weld material and reactor vessel design data.

3.1 J-Integral Resistance Model for Mn-Mo-Ni/Linde 80 Welds

A model for the *J*-integral resistance versus crack extension curve (*J-R* curve) required to analyze low upper-shelf energy materials has been derived specifically for Mn-Mo-Ni/Linde 80 weld materials. A previous analysis of the reactor vessels of B&W Owners Group RVWG [1] described the development of this toughness model from a large data base of fracture specimens. A lower bound ($-2S_e$) *J-R* curve is obtained by multiplying *J*-integrals from the mean *J-R* curve by 0.699 [1]. It was shown in a previous low upper-shelf toughness analysis performed for B&W Owners Group plants [6] that a typical lower bound *J-R* curve is a conservative representation of toughness values for reactor vessel beltline materials, as required by Appendix K [4] for Levels A, B, and C Service Loadings. The best estimate representation of toughness required for Level D Service Loadings is provided by the mean *J-R* curve [7].

3.2 Reactor Vessel Design Data

Pertinent design data for upper-shelf flaw evaluations in the beltline region of the reactor vessel are provided below for Point Beach Units 1 and 2.

Design Pressure, P_d	= 2485 psig [2] (use 2500 psig)
Inside radius, R_i	= 66 in. [2]
Vessel thickness, t	= 6.5 in. [2]
Nominal cladding thickness, t_c	= 0.1875 in. [2]

3.3 Mechanical Properties for Weld Material

Mechanical properties for the base and weld materials are presented in Tables 3-1 through 3-3. The reactor vessel base metal at Point Beach Unit 1 is SA-302, Grade B low alloy steel, and at Point Beach Unit 2 is SA-508, Grade 2, Class 1 low alloy steel [8]. Base metal properties are found in the ASME Code [9]. Weld metal tensile properties are taken from appropriate surveillance capsule data of each weld material. The ASME* transition region fracture toughness curve for K_{Ic}, used to define the beginning of the upper-shelf toughness region, is indexed by the initial RT_{NDT} of the weld material. Also, Poisson's ratio, ν, is taken to be 0.3.

3.3.1 Axial Weld SA-847

Table 3-1 Mechanical Properties for SA-847 Weld of Point Beach Unit 1

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-847	Base Metal	Weld SA-847	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [10]	Code [9]	Actual [10]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	29200	50.00	95.00	80	99.8	7.06E-06
200	28500	47.50	89.60	80	99.8	7.25E-06
300	28000	46.10	86.01	80	99.8	7.43E-06
335	27790	45.74	85.10	80	97.6	7.48E-06
400	27400	45.10	84.77	80	99.8	7.58E-06
500	27000	44.50	84.26	80	99.8	7.70E-06
541.4	26751.6	44.16	84.04	80	99.8	7.75E-06
544.5	26733	44.14	84.03	80	99.8	7.76E-06
550	26700	44.11	84.00	80	99.8	7.77E-06
600	26400	43.80	83.74	80	99.8	7.83E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = -5.0°F [5]
Margin = 48.3°F [5]

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3.3.2 Circumferential Weld SA-1011

Table 3-2 Mechanical Properties for SA-1101 Weld of Point Beach Unit 1

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-1101	Base Metal	Weld SA-1101	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [11]	Code [9]	Actual [11]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	29200	50.00	93.66	80	105.10	7.06E-06
200	28500	47.50	92.20	80	104.90	7.25E-06
300	28000	46.10	90.74	80	104.70	7.43E-06
400	27400	45.10	89.29	80	104.50	7.58E-06
500	27000	44.50	87.83	80	104.30	7.70E-06
541.4	26751.6	44.14	87.23	80	104.21	7.76E-06
544.5	26733	44.14	87.18	80	104.21	7.76E-06
550	26700	44.11	87.10	80	104.20	7.77E-06
600	26400	43.80	86.37	80	104.10	7.83E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = 10.0°F [5]
Margin = 56.0°F [5]

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3.3.3 Circumferential Weld SA-1484

Table 3-3 Mechanical Properties for SA-1484 Weld of Point Beach Unit 2

Temp.	E	Yield Strength (σ_y)		Ultimate Strength (σ_u)*		α
Material:	Base Metal	Base Metal	Weld SA-1484	Base Metal	Weld SA-1484	Base Metal
Source: [Ref.]	Code [9]	Code [9]	Actual [12]	Code [9]	Actual [12]	Code [9]
(°F)	(ksi)	(ksi)	(ksi)	(ksi)	(ksi)	(in/in/°F)
100	27800	50.00	82.10	80	96.90	6.50E-06
200	27100	47.50	79.57	80	92.98	6.67E-06
300	26700	46.10	78.00	80	90.40	6.87E-06
400	26100	45.10	77.17	80	89.41	7.07E-06
450	25900	44.76	76.80	80	89.60	7.15E-06
500	25700	44.50	76.42	80	90.29	7.25E-06
541.4	25460	44.16	76.15	80	91.25	7.32E-06
544.5	25444	44.14	76.13	80	91.34	7.33E-06
580	25264	43.94	76.00	80	92.50	7.39E-06
600	25200	43.80	75.80	80	93.28	7.42E-06

* Note: The ultimate strength values of the base and weld metals given here are not used in calculations

Initial RT_{NDT} = -5.0°F [5]
Margin = 68.5°F [5]

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4.0 Analytical Methodology

Upper-shelf toughness is evaluated through use of fracture mechanics analytical methods that utilize the acceptance criteria and evaluation procedures of Section XI, Appendix K [4], where applicable.

4.1 Procedure for Evaluating Levels A and B Service Loadings

The applied *J*-integral is calculated per Appendix K, paragraph K-4210 [4], using an effective flaw depth to account for small scale yielding at the crack tip, and evaluated per K-4220 for upper-shelf toughness and per K-4310 for flaw stability.

4.2 Procedure for Evaluating Levels C and D Service Loadings

Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of the PCRIT computer code to determine stress intensity factors. The beltline region welds identified in Section 3.3 are analyzed for all Level C and D transients. Two Level D transients are specified for the Point Beach Units. The original equipment specification includes a Steam Line Break (SLB) transient and a Reactor Coolant Line Break (LOCA) transient. The Point Beach FSAR contains a Steam Line Break (two loops in service) without Offsite Power transient [13].

The transients considered appear in Figure 5.1. Transients are assumed to hold steady at the end of their definitions, and are held constant until the thermal gradient through the shell has developed fully and begins to dissipate.

The evaluation is performed as follows:

(1) For each transient described above, utilize PCRIT to calculate stress intensity factors for a semi-elliptical flaw of depth $^1/_{10}$ of the base metal wall thickness, as a function of time, due to internal pressure and radial thermal gradients with a factor of safety of 1.0 on loading. The applied stress intensity factor, K_I, calculated by PCRIT for each of these transients is compared to the K_{Jc} limit of the weld. The transient that most closely approaches the K_{Jc} limit is chosen as the limiting transient, and the critical time in the limiting transient occurs at the point where K_I most closely approaches the upper-shelf toughness curve.

(2) At the critical transient time, develop a crack driving force diagram with the applied *J*-integral and *J-R* curves plotted as a function of flaw extension. The adequacy of the upper-shelf toughness is evaluated by comparing the applied *J*-integral with the *J-R* curve at a flaw extension of 0.10 in. Flaw stability is assessed by examining the slopes of the applied *J*-integral and *J-R* curves at the points of intersection.

(3) Verify that the extent of stable flaw extension is no greater than 75% of the vessel wall thickness by determining when the applied *J*-integral curve intersects the mean *J-R* curve.

(4) Verify that the remaining ligament is not subject to tensile instability. The internal pressure p shall be less than P_l, where P_l is the internal pressure at tensile instability of the remaining ligament. Equations for P_l are given below for the axial and circumferential flaws [14]. These equations first appear in the 2001 Edition of the ASME Section XI code that is cited.

(a) For an axial flaw,

$$P_l = 1.07\sigma_O\left[\frac{1-(A_c/A)}{(R_i/t)+(A_c/A)}\right] \qquad \text{[eqn. 1]}$$

where

$$\sigma_O = \frac{\sigma_y + \sigma_u}{2} \qquad \text{[eqn. 2]}$$

$$A = t(\ell + t) \qquad \text{[eqn. 3]}$$

$$A_c = \frac{\pi a \ell}{4} \qquad \text{[eqn. 4]}$$

and

ℓ = surface length of crack, six times the depth, a
R_m = mean radius of vessel

This equation for P_l includes the effect of pressure on the flaw face.

(b) For a circumferential flaw,

$$P_l = 1.07\sigma_O\left[\frac{1-(A_c/A)}{(R_i^2/(2R_m t))+(A_c/A)}\right] \qquad \text{[eqn. 5]}$$

where σ_O, A, and A_c are given by equations 2, 3 and 4, respectively.

This equation for P_l includes the effect of pressure on the flaw face. This equation is valid for internal pressures not exceeding the pressure at tensile instability caused by the applied hoop stress acting over the nominal wall thickness of the vessel. This validity limit on pressure for the circumferential flaw equation for P_l is

$$P_l \le 1.07\sigma_O\left[\frac{t}{R_i}\right] \qquad \text{[eqn. 6]}$$

4.3 Temperature Range for Upper-Shelf Fracture Toughness Evaluations

Upper-shelf fracture toughness is determined through use of Charpy V-notch impact energy versus temperature plots by noting the temperature above which the Charpy energy remains on a plateau, maintaining a relatively high constant energy level. Similarly, fracture toughness can be addressed in three different regions on the temperature scale, i.e. a lower-shelf toughness region, a transition region, and an upper-shelf toughness region. Fracture toughness of reactor vessel steel and associated weld metals are conservatively predicted by the ASME initiation toughness curve, K_{Ic}, in the lower-shelf and transition regions. In the upper-shelf region, the upper-shelf toughness curve, K_{Jc}, is derived from the upper-shelf *J*-integral resistance model described in Section 3.1. The upper-shelf toughness then becomes a function of fluence, copper content, temperature, and fracture specimen size. When upper-shelf toughness is plotted versus temperature, a plateau-like curve develops that decreases slightly with increasing temperature. Since the present analysis addresses the low upper-shelf toughness issue, only the upper-shelf temperature range, which begins at the intersection of K_{Ic} and the upper-shelf toughness curves, K_{Jc}, is considered.

4.4 Effect of Cladding Material

The PCRIT code utilized in the flaw evaluations for Levels C and D Service Loadings does not consider stresses in the cladding when calculating stress intensity factors for thermal loads. To account for this cladding effect, an additional stress intensity factor, K_{Iclad}, is calculated separately and added to the total stress intensity factor computed by PCRIT.

The contribution of cladding stresses to stress intensity factor was examined previously [2]. In this low upper-shelf toughness analysis performed for B&W Owners Group Reactor Vessel Working Group plants, the Zion-1 WF-70 weld using thermal loads from the Turkey Point SLB was determined to be the bounding case. The Zion-1 vessel was as thick as or thicker than any other vessel. The thicknesses of the reactor vessels for the both Point Beach units are 6.5" whereas the Zion vessel is 8.44". The nominal cladding thickness is 3/16" for both vessels. From a thermal stress perspective, it is conservative to consider the thicker vessel. For the Zion vessel, the maximum value of K_{Iclad}, at any time during the transient and for any flaw depth, was determined to be 9.0 ksi$\sqrt{\text{in}}$. This bounding value is therefore used as the stress intensity factor for K_{Iclad} in this Point Beach low upper-shelf toughness analysis.

5.0 Applied Loads

The Levels A and B Service Loadings required by Appendix K are an accumulation pressure (internal pressure load) and a cooldown rate (thermal load). Since Levels C and D Service Loadings are not specified by the Code, Levels C and D pressurized thermal shock events are reviewed and a worst case transient is selected for use in flaw evaluations.

5.1 Levels A and B Service Loadings

Per paragraph K-1300 of Appendix K [4], the accumulation pressure used for flaw evaluations should not exceed 1.1 times the design pressure. Using 2.5 ksi as the design pressure, the accumulation pressure is 2.75 ksi. The cooldown rate is also taken to be the maximum required by Appendix K, 100°F/hour.

5.2 Levels C and D Service Loadings

As discussed in Section 4.2, the SLB and LOCA transients are evaluated using the computer code PCRIT. Pressure and temperature time histories for the two transients considered are shown in Figure 5-1.

Figure 5-1 Level D transients – Reactor Coolant Temperature and Pressure vs. Time

[ ]

6.0 Evaluation for Levels A and B Service Loadings

The material mean and lower bounding *J-R* values for Evaluation Conditions 1, 2 and 3 detailed in Table 2-1 are given in Tables 6-1 through 6-3, respectively. Initial flaw depths equal to $^1/_4$ of the vessel wall thickness are analyzed for Levels A and B Service Loadings following the procedure outlined in Section 4.1 and evaluated for acceptance based on values for the *J*-integral resistance of the materials from Section 3.3. The results of the evaluation are presented in Table 6-4 through 6-6, where it is seen that the minimum ratio of material *J*-integral resistance ($J_{0.1}$) to applied *J*-integral (J_1) is 1.87 for the SA-847 axial weld for Evaluation Condition 2, current power conditions without hafnium power suppression assemblies. This ratio is higher than the minimum acceptable value of 1.0. Also included in Table 6-4 through 6-6 is the applied *J*-integral at ($J_{0.1}$) with a safety factor on pressure of 1.25.

The flaw evaluation for the controlling weld (SA-847) and controlling Evaluation Condition (2) is repeated by calculating applied *J*-integrals for various amounts of flaw extension with safety factors (on pressure) of 1.15 and 1.25. The results, along with mean and lower bound *J-R* curves, are plotted in Figure 6-1. The requirement for ductile and stable crack growth is also demonstrated by Figure 6-1 since the slope of the applied *J*-integral curve for a safety factor of 1.25 is considerably less than the slope of the lower bound *J-R* curve at the point where the two curves intersect.

Table 6-1 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld Material ID	Controlling Weld Weld Orientation	Controlling Weld Cu Content (wt%)	Fluence × 10^18 (n/cm²) at I.S.	Fluence × 10^18 (n/cm²) at t/4	*J-R* at Δa = 0.1 in. Mean (lb/in)	*J-R* at Δa = 0.1 in. Lower Bound at -2Se (lb/in)
PB-1	541.4	SA-847	L	0.23	32.45	21.45	886	619
PB-1	541.4	SA-1101	C	0.23	47.10	31.13	871	609
PB-2	541.4	SA-1484	C	0.26	48.45	32.03	828	579

Table 6-2 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld Material ID	Controlling Weld Weld Orientation	Controlling Weld Cu Content (wt%)	Fluence × 10^18 (n/cm²) at I.S.	Fluence × 10^18 (n/cm²) at t/4	*J-R* at Δa = 0.1 in. Mean (lb/in)	*J-R* at Δa = 0.1 in. Lower Bound at -2Se (lb/in)
PB-1	544.5	SA-847	L	0.23	31.15	20.59	885	618
PB-1	544.5	SA-1101	C	0.23	45.20	29.88	870	608
PB-2	544.5	SA-1484	C	0.26	46.45	30.70	827	578

Table 6-3 Material *J*-Integral Resistance for Levels A and B Service Loadings – Evaluation Condition 3 – Current Power Conditions With Hafnium Assemblies

Plant	Cold Leg Temp. (°F)	Controlling Weld Material ID	Controlling Weld Weld Orientation	Controlling Weld Cu Content (wt%)	Fluence × 10^18 (n/cm²) at I.S.	Fluence × 10^18 (n/cm²) at t/4	*J-R* at Δa = 0.1 in. Mean (lb/in)	*J-R* at Δa = 0.1 in. Lower Bound at -2Se (lb/in)
PB-1	544.5	SA-847	L	0.23	26.65	17.62	891	623
PB-1	544.5	SA-1101	C	0.23	38.20	25.25	877	613
PB-2	544.5	SA-1484	C	0.26	37.85	25.02	836	585

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Table 6-4 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 1 – Uprated Power Conditions Without Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	619	331	1.87	388	1.60
PB-1	SA-1101	C	609	98	6.21	113	5.39
PB-2	SA-1484	C	579	104	5.57	119	4.87

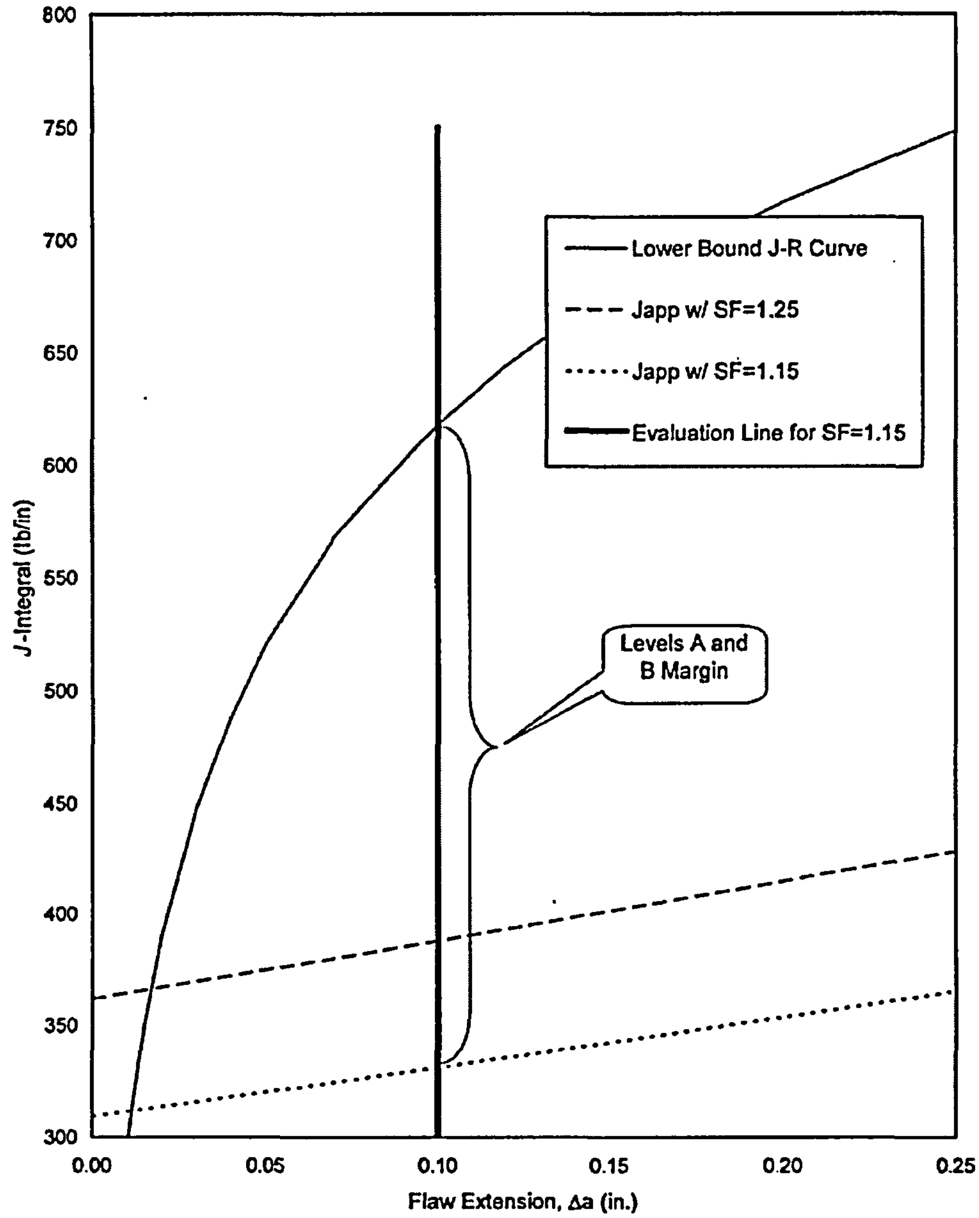
Table 6-5 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 2 – Current Power Conditions Without Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	618	331	1.87	388	1.59
PB-1	SA-1101	C	608	98	6.20	113	5.38
PB-2	SA-1484	C	578	104	5.56	119	4.86

Table 6-6 Flaw Evaluation for Levels A and B Service Loadings – Evaluation Condition 3 – Current Power Conditions With Hafnium Assemblies

Plant	Weld Number	Weld Orientation	Lower Bounding $J_{0.1}$ at t/4 (lb/in)	SF = 1.15		SF = 1.25	
				J_1 (lb/in)	$J_{0.1}/J_1$	J_1 (lb/in)	$J_{0.1}/J_1$
PB-1	SA-847	L	623	331	1.88	388	1.61
PB-1	SA-1101	C	613	98	6.26	113	5.42
PB-2	SA-1484	C	585	104	5.63	119	4.92

Figure 6-1 J-Integral vs. Flaw Extension for Levels A & B Service Loadings - Evaluation Condition 2 - Current Power Conditions Without Hafnium Assemblies - Weld SA-847



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7.0 Evaluation for Levels C and D Service Loadings

A flaw depth of $^1/_{10}$ of the base metal wall thickness, plus the cladding thickness, is used to evaluate the Level D Service Loadings. The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual stress, deadweight, and pressure terms. PCRIT is run for each Level D transient. RT_{NDT} is also calculated by PCRIT. Transition region toughness is obtained from the ASME Section XI equation for crack initiation [15].

$$K_{Ic} = 33.2 + 2.806 \exp[0.02(T - RT_{NDT} + 100°F)] \qquad \text{[eqn. 7]}$$

where:

K_{Ic} = transition region toughness, ksi$\sqrt{\text{in}}$
T = crack tip temperature, °F

Upper-shelf toughness is derived from the J-integral resistance model of Section 3.1 for a flaw depth of $^1/_{10}$ of the wall thickness, a crack extension of 0.10 in., and fluence, as follows:

$$K_{Jc} = \sqrt{\frac{J_{0.1}E}{1000(1-v^2)}} \qquad \text{[eqn. 8]}$$

where

K_{Jc} = upper-shelf region toughness, ksi$\sqrt{\text{in}}$
$J_{0.1}$ = J-integral resistance at Δa = 0.1 in.

Figure 7-1 through 7-3 shows the variation of applied stress intensity factor, K_I, transition range toughness, K_{Ic}, and upper-shelf toughness, K_{Jc} with temperature for the Evaluation Condition 1 described in Table 2-1 for the three welds. The markers on the K_I curve indicate points in time at which PCRIT solutions are available. For all the three welds that were analyzed, the LOCA transient is limiting since it most closely approaches the K_{Jc} limit of each weld. All subsequent analysis will pertain to this transient. In the upper-shelf toughness range, the K_I curve is closest to the lower bound K_{Jc} curve at a particular time point into the transient for each weld, as listed below:

Weld	Time (min)
SA-847	2.40
SA-1011	1.50
SA-1484	1.30

For each weld, the time specified above is selected as the critical time in the transient at which to perform the flaw evaluation for Level D Service Loadings.

Figure 7-1 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-847

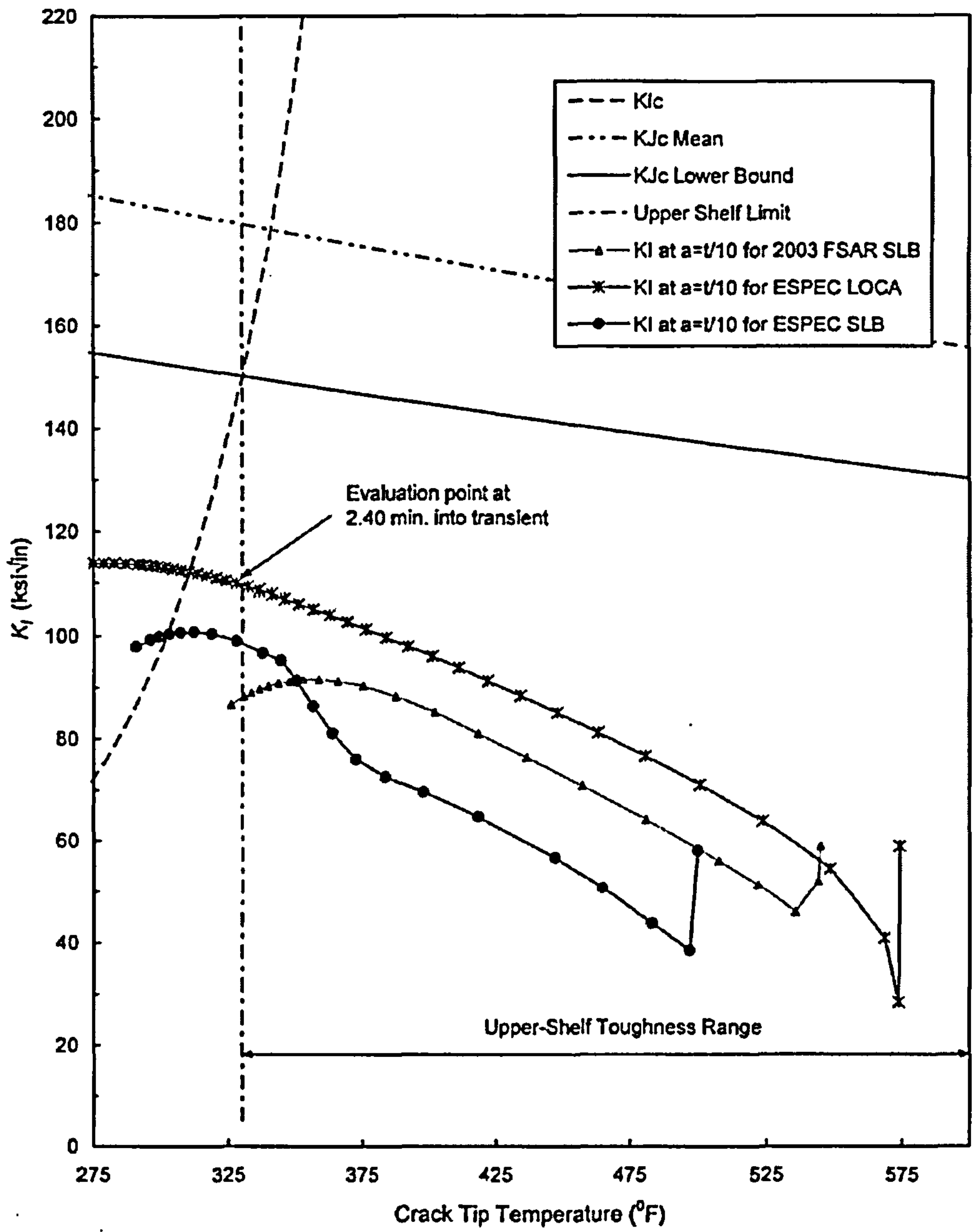

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Figure 7-2 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1101

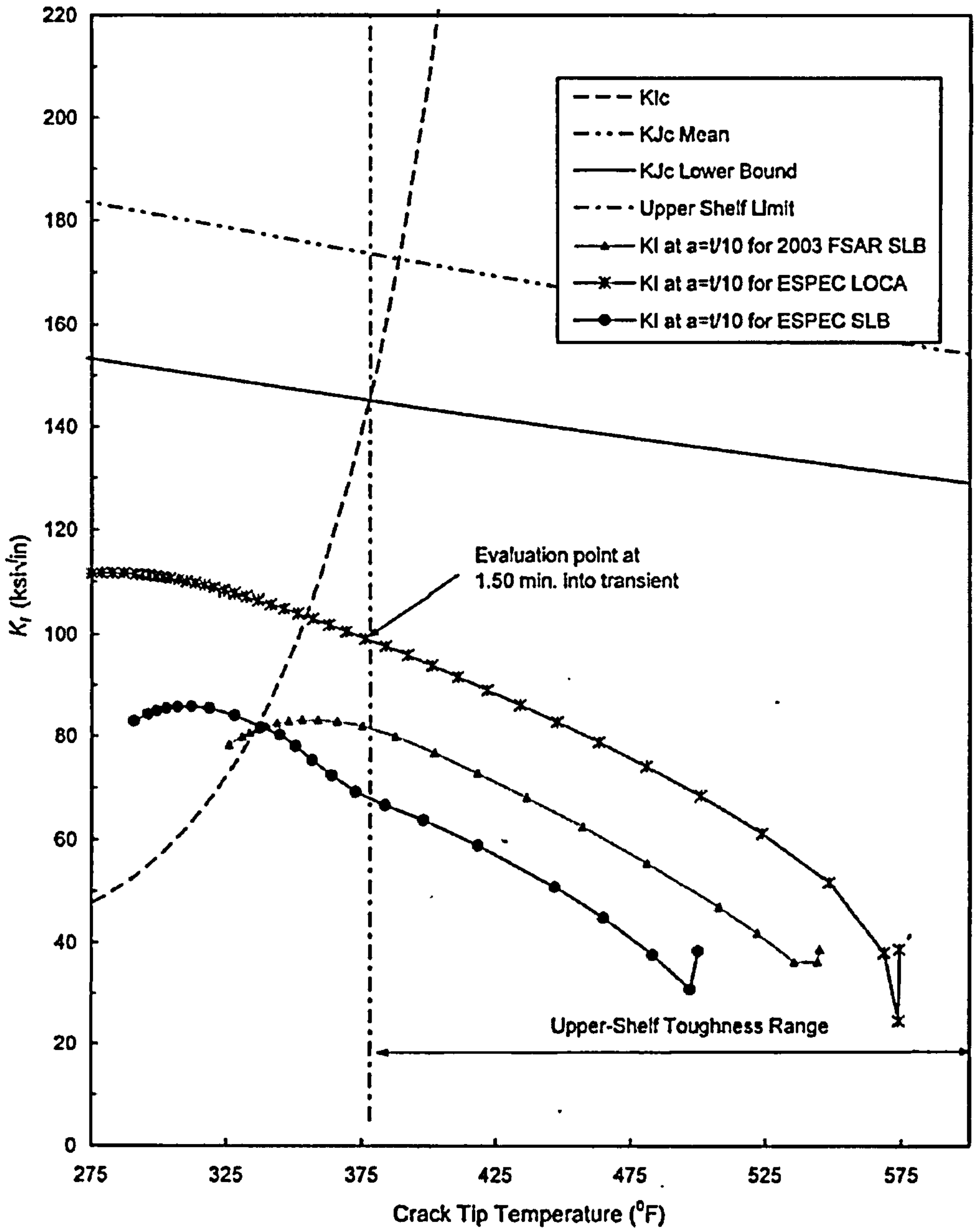

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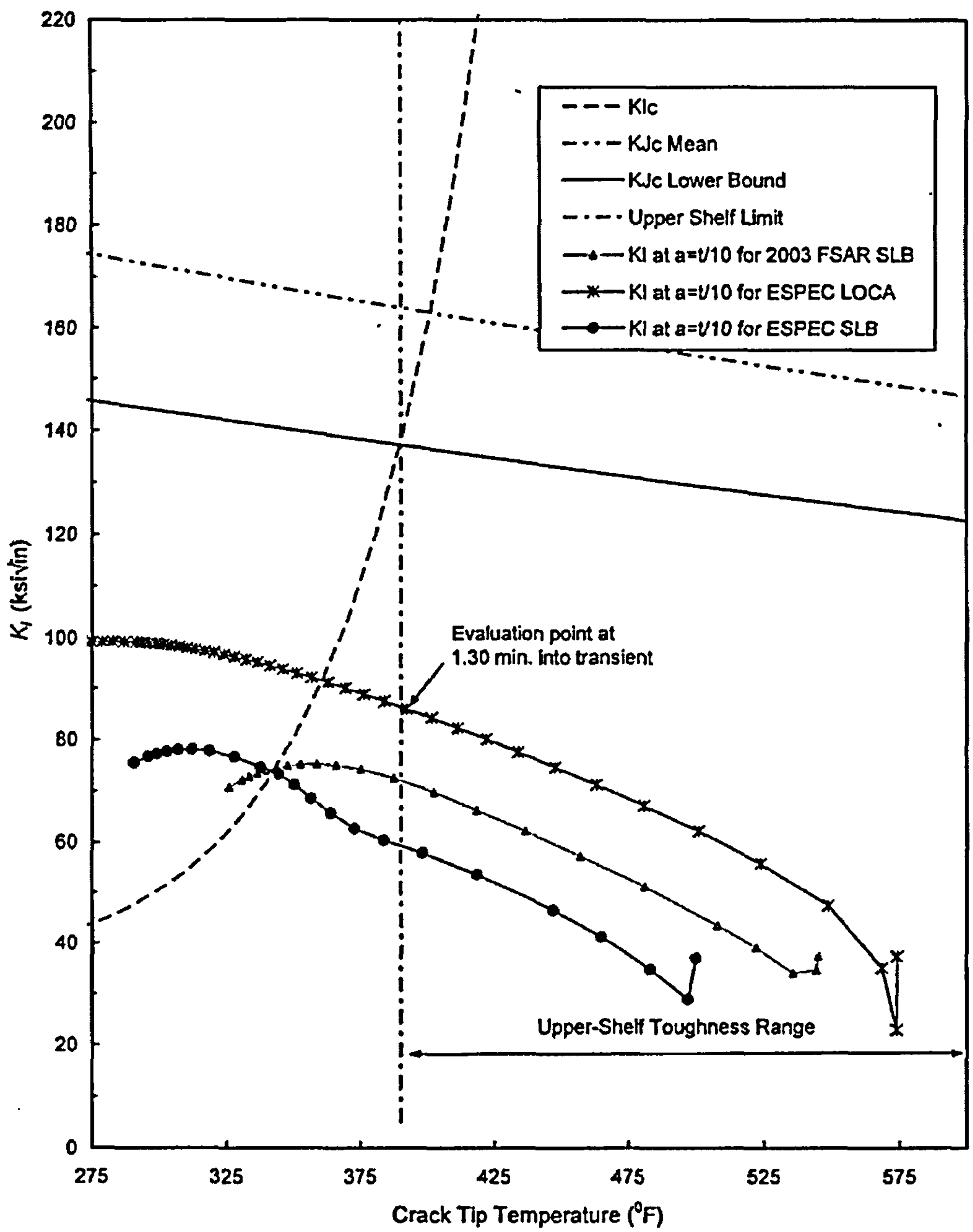
Figure 7-3 K_I vs. Crack Tip Temperature for Evaluation Condition 1 - SA-1484



Applied *J*-integrals for the LOCA transient are calculated for each weld at the critical time points identified above for various flaw depths in Table 7-1, 7-2, and 7-3 using stress intensity factors from PCRIT and adding 9.0 ksi$\sqrt{\text{in}}$ to account for cladding effects. Stress intensity factors are converted to *J*-integrals by the plain strain relationship,

$$J_{applied}(a) = 1000\frac{K_{Itotal}^2(a)}{E}(1-\nu^2) \qquad \text{[eqn. 9]}$$

Tables 7-1, 7-2, and 7-3 lists flaw extensions vs. applied *J*-integrals. As the Point Beach vessels are 6.5 in. thick, the initial flaw depth of $^1/_{10}$ of the wall thickness is 0.65 in. Flaw extension from this flaw depth is calculated by subtracting 0.65 in. from the built-in PCRIT flaw depths in the base metal. The results, along with mean *J-R* curve, are plotted in Figure 7-4. This figure indicates that Weld SA-847 is limiting as the ratio of the applied *J*-integral to the material *J-R* curve is less than the other two welds. Figure 7-5 is a plot of the applied *J*-integrals and the mean *J-R* curves for the three Evaluation Conditions from Table 2-1 for Weld SA-847. Evaluation Condition 1, uprated power conditions without hafnium power suppression assemblies, is the limiting case as the ratio of the mean *J-R* curves to applied *J*-integrals is the minimum of the three Evaluation Conditions. The requirements for ductile and stable crack growth are demonstrated by Figure 7-5 since the slopes of the applied *J*-integral curves are considerably less than the slopes mean *J-R* curves at the points of intersection. The Level D Service Loading requirement that the extent of stable flaw extension be no greater than 75% of the vessel wall thickness is easily satisfied since the applied *J*-integral curves intersects the mean *J-R* curves at flaw extensions that are only a small fraction of the wall thickness (less than 1%).

The last requirement is that the internal pressure *p* shall be less than P_l, the internal pressure at tensile instability of the remaining ligament. Table 7-4 gives the results of the calculations for P_l for flaw depths up to 1.365 inches for Evaluation Condition 1. As the internal pressure *p* is less than P_l, the remaining ligament is not subject to tensile instability.

Table 7-1 J-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-847

Time =	2.40 min				E =	26751.6	ksi
Crack tip at t/10		t =	6.5	in.	ν =	0.3	
$(a^{++}/t)*40$	a^{++} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		246.40	62.06	9.0	71.1	172
2	0.3250		274.80	83.65	9.0	92.7	292
3	0.4875		302.10	94.64	9.0	103.6	365
4	0.6500	0.0000	328.00	100.97	9.0	110.0	411
5	0.8125	0.1625	352.70	104.24	9.0	113.2	436
6	0.9750	0.3250	375.90	105.82	9.0	114.8	448
7	1.1375	0.4875	397.70	106.12	9.0	115.1	451
8	1.3000	0.6500	417.90	105.76	9.0	114.8	448
9	1.4625	0.8125	436.50	104.86	9.0	113.9	441
10	1.6250	0.9750	453.60	103.22	9.0	112.2	428
12	1.9500	1.3000	483.10	98.74	9.0	107.7	395
14	2.2750	1.6250	507.00	93.05	9.0	102.1	354
16	2.6000	1.9500	525.80	88.28	9.0	97.3	322
18	2.9250	2.2750	540.10	82.87	9.0	91.9	287
20	3.2500	2.6000	550.70	77.27	9.0	86.3	253
22	3.5750	2.9250	558.40	71.71	9.0	80.7	222
24	3.9000	3.2500	563.90	66.53	9.0	75.5	194
26	4.2250	3.5750	567.60	61.81	9.0	70.8	171
28	4.5500	3.9000	570.00	57.20	9.0	66.2	149
30	4.8750	4.2250	571.60	52.58	9.0	61.6	129
32	5.2000	4.5500	572.60	48.13	9.0	57.1	111

Note: a^{++} is the flaw depth in the base metal

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Table 7-2 *J*-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-1101

Time =	1.50 min				E =	26751.6	ksi
Crack tip at t/10		t =	6.5	in.	ν =	0.3	
$(a^{**}/t)*40$	a^{**} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		280.80	59.65	9.0	68.7	160
2	0.3250		314.80	78.57	9.0	87.6	261
3	0.4875		346.70	86.65	9.0	95.7	311
4	0.6500	0.0000	376.30	90.22	9.0	99.2	335
5	0.8125	0.1625	403.60	91.26	9.0	100.3	342
6	0.9750	0.3250	428.40	90.74	9.0	99.7	338
7	1.1375	0.4875	450.60	89.06	9.0	98.1	327
8	1.3000	0.6500	470.50	86.71	9.0	95.7	312
9	1.4625	0.8125	488.00	83.66	9.0	92.7	292
10	1.6250	0.9750	503.10	80.42	9.0	89.4	272
12	1.9500	1.3000	527.20	72.98	9.0	82.0	229
14	2.2750	1.6250	544.30	65.06	9.0	74.1	187
16	2.6000	1.9500	555.90	57.27	9.0	66.3	149
18	2.9250	2.2750	563.40	49.24	9.0	58.2	115
20	3.2500	2.6000	568.10	41.31	9.0	50.3	86
22	3.5750	2.9250	570.90	34.09	9.0	43.1	63
24	3.9000	3.2500	572.40	27.47	9.0	36.5	45
26	4.2250	3.5750	573.30	21.94	9.0	30.9	33
28	4.5500	3.9000	573.70	17.63	9.0	26.6	24
30	4.8750	4.2250	573.90	14.36	9.0	23.4	19
32	5.2000	4.5500	574.00	11.59	9.0	20.6	14

Note: a^{**} is the flaw depth in the base metal

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Table 7-3 *J*-Integral vs. Flaw Extension for Evaluation Condition 1 - SA-1484

Time =	1.30 min				E =	25459.9	ksi
Crack tip at t/10		t =	6.5	in.	ν =	0.3	
$(a^{**}/t)*40$	a^{**} (in.)	Δa (in.)	Temp. (F)	K_{Isum}	K_{Iclad}	K_{Itotal}	J_{app} (lb/in)
1	0.1625		292.60	51.19	9.0	60.2	129
2	0.3250		328.30	67.16	9.0	76.2	207
3	0.4875		361.60	73.97	9.0	83.0	246
4	0.6500	0.0000	392.10	76.91	9.0	85.9	264
5	0.8125	0.1625	419.80	77.72	9.0	86.7	269
6	0.9750	0.3250	444.70	77.16	9.0	86.2	265
7	1.1375	0.4875	466.60	75.59	9.0	84.6	256
8	1.3000	0.6500	485.80	73.43	9.0	82.4	243
9	1.4625	0.8125	502.50	70.67	9.0	79.7	227
10	1.6250	0.9750	516.40	67.71	9.0	76.7	210
12	1.9500	1.3000	538.10	61.07	9.0	70.1	175
14	2.2750	1.6250	552.60	54.04	9.0	63.0	142
16	2.6000	1.9500	561.80	47.18	9.0	56.2	113
18	2.9250	2.2750	567.40	40.21	9.0	49.2	87
20	3.2500	2.6000	570.60	33.42	9.0	42.4	64
22	3.5750	2.9250	572.40	27.38	9.0	36.4	47
24	3.9000	3.2500	573.30	21.99	9.0	31.0	34
26	4.2250	3.5750	573.80	17.69	9.0	26.7	25
28	4.5500	3.9000	574.00	14.53	9.0	23.5	20
30	4.8750	4.2250	574.00	12.34	9.0	21.3	16
32	5.2000	4.5500	574.10	10.58	9.0	19.6	14

Note: a^{**} is the flaw depth in the base metal

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Table 7-4 Level D Service Loadings - Internal Pressure at Tensile Instability - SA-847

flaw depth a (in.)	P_i (ksi)
0.0650	9.18
0.1300	9.16
0.1950	9.14
0.2600	9.12
0.3250	9.09
0.3900	9.06
0.4550	9.02
0.5200	8.98
0.5850	8.93
0.6500	8.88
0.7150	8.84
0.7800	8.78
0.8450	8.73
0.9100	8.68
0.9750	8.62
1.0400	8.56
1.1050	8.51
1.1700	8.45
1.2350	8.39
1.3000	8.32
1.3650	8.26

Figure 7-4. *J*-Integral vs. Flaw Extension – All Welds

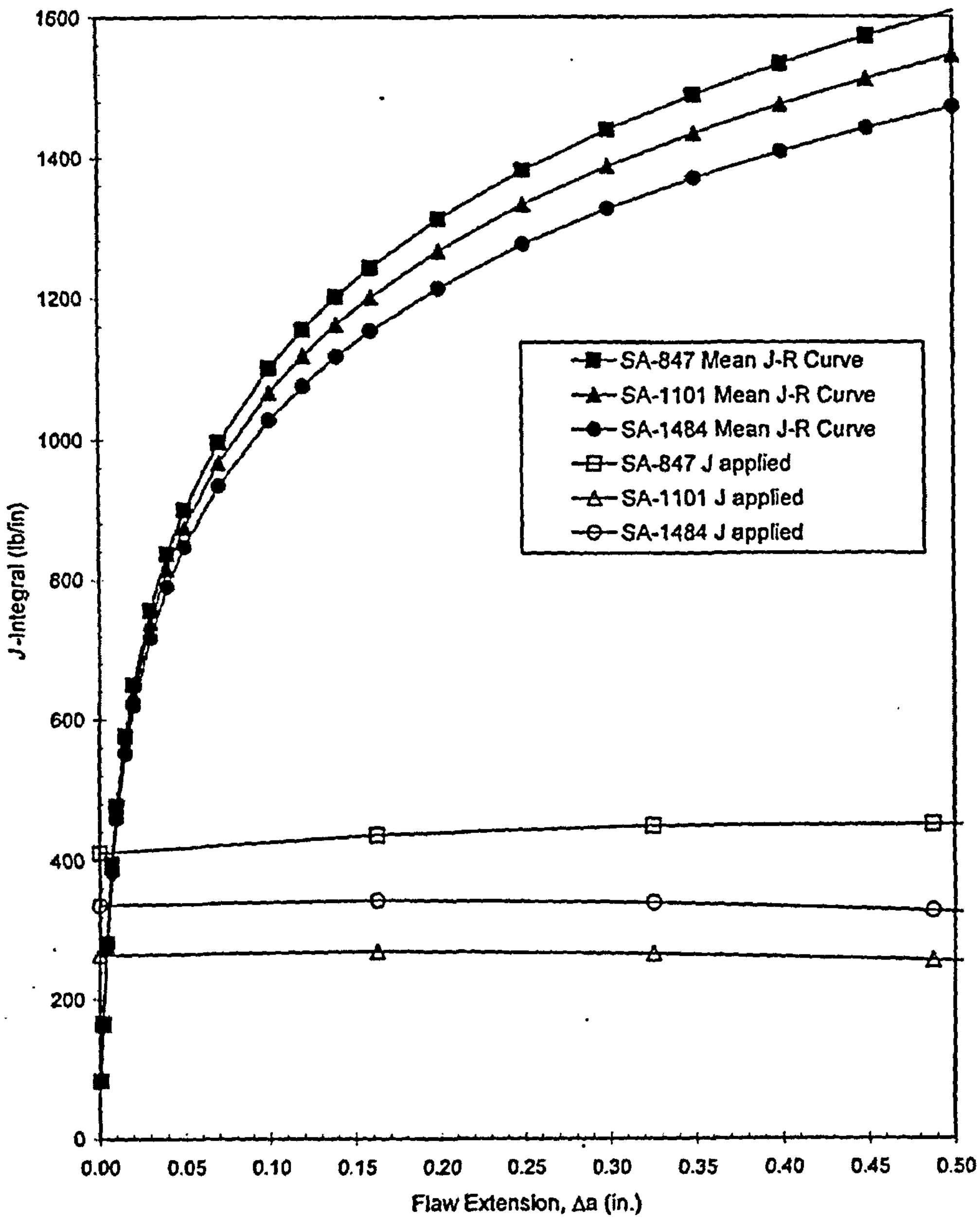
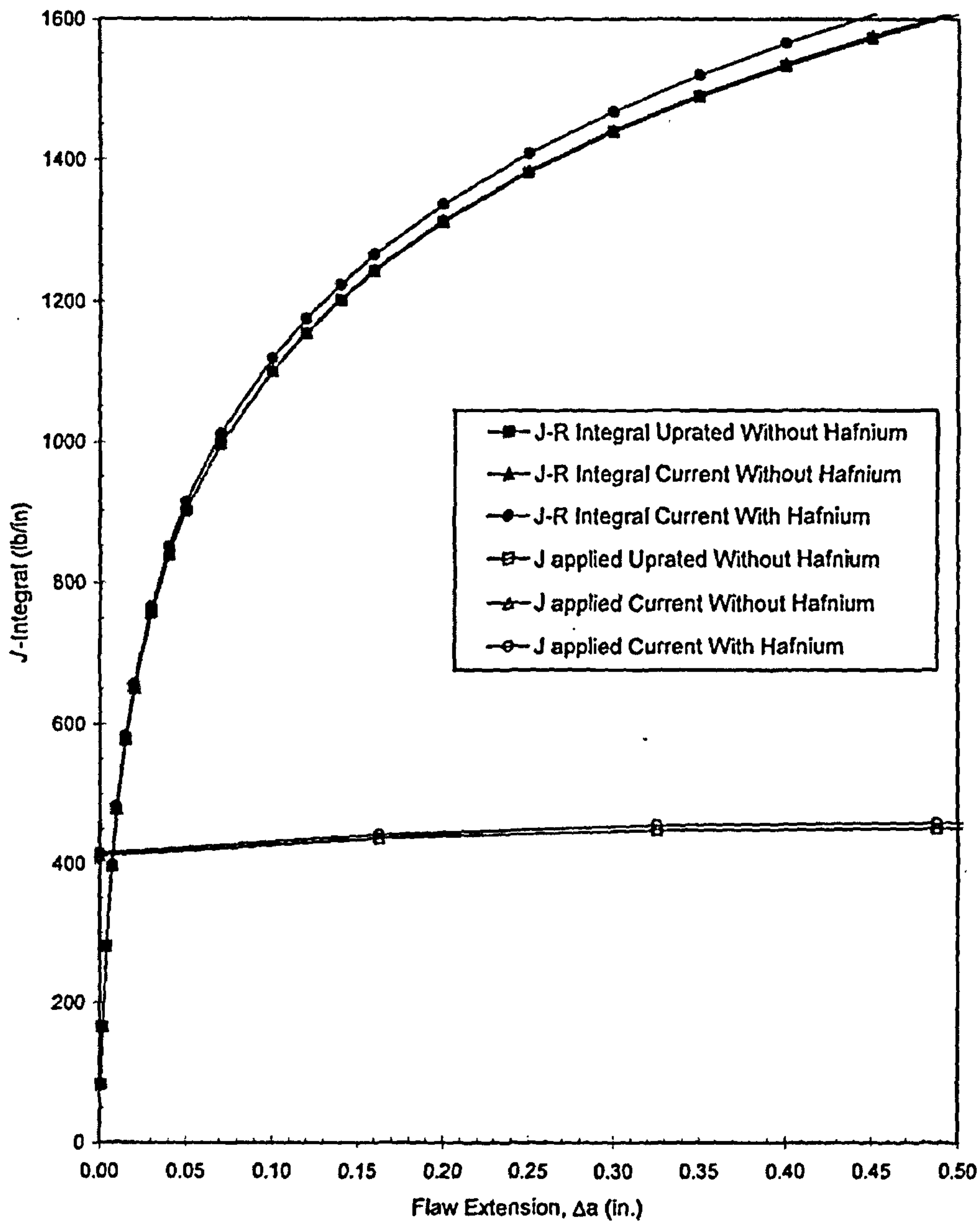


Figure 7-5. *J*-Integral vs. Flaw Extension – Weld SA-847



8.0 Summary of Results

A low upper-shelf toughness fracture mechanics analysis has been performed to evaluate the reactor vessel welds at Point Beach Units 1 and 2 for projected low upper-shelf energy levels at 53 EFPY, considering Levels A, B, C, and D Service Loadings of the ASME Code.

Evidence that the ASME Code, Section XI, Appendix K [4] acceptance criteria have been satisfied for Levels A and B Service Loadings is provided by the following:

(1) The limiting weld is the axial weld SA-847 of Point Beach Unit 1 in the current power condition without hafnium power suppression assemblies. Figure 6-1 shows that with factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied *J*-integral (J_1) is less than the *J*-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ = 1.87 which is significantly greater than the required value of 1.0.

(2) Figure 6-1 shows that with a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied *J*-integral curve is less than the slope of the lower bound *J-R* curve at the point where the two curves intersect.

Evidence that the ASME Code, Section XI, Appendix K [4] acceptance criteria have been satisfied for Level D Service Loadings is provided by the following:

(1) Figure 7-5 shows that with a factor of safety of 1.0 on loading, flaw extensions are ductile and stable since the slope of the applied *J*-integral curve is less than the slopes of both the lower bound and mean *J-R* curves at the points of intersection.

(3) Figure 7-5 shows that the flaw remains stable at much less than 75% of the vessel wall thickness. It has also been shown that the remaining ligament is sufficient to preclude tensile instability by a large margin.

9.0 Conclusion

The limiting Point Beach Units 1 and 2 reactor vessel beltline weld (axial weld SA-847 of Unit 1) satisfies the acceptance criteria of Appendix K to Section XI of the ASME Code [4] for projected low upper-shelf Charpy impact energy levels at 53 effective full power years of plant operation for the three conditions evaluated: uprated power conditions (1678 MWt) without hafnium power suppression assemblies, current power conditions (1540 MWt) without hafnium power suppression assemblies, and current power conditions (1540 MWt) with hafnium power suppression assemblies.

10.0 References

1. BAW-2192PA, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A & B Service Loads, April 1994.

2. BAW-2178PA, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C & D Service Loads, April 1994.

3. BAW-2255, Effect of Power Upgrade on Low Upper-Shelf Toughness Issue, May 1995.

4. ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition with Addenda through 2000.

5. USNRC Reactor Vessel Integrity Database Version 2.0.1 (RVID).

6. BAW-2275, Low Upper-Shelf Toughness Fracture Mechanics Analysis of B&W Designed Reactor Vessels for 48 EFPY, August 1996.

7. BAW-2312, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life through 48 Effective Full Power Years, December 2000.

8. BAW-2150, Materials Information for Westinghouse-Designed Reactor Vessels Fabricated by B&W, December 1990.

9. ASME Boiler and Pressure Vessel Code, Section III, Appendices, 1989 Edition with no Addenda.

10. WCAP-13902, Analysis of Capsule S from the Rochester Gas and Electric Corporation R. E. Ginna Reactor Vessel Radiation Surveillance Program, December 1993.

11. WCAP-15916, Analysis of Capsule X from the Florida Power and Light Turkey Point 3 Reactor Vessel Radiation Surveillance Program, September 2002.

12. BAW-2254, Test Results of Capsule CR3-LG2: B&W Owners Group – Master Integrated Reactor Vessel Surveillance Program, October 1995.

13. Point Beach Nuclear Plant Units 1 and 2 Final Safety Analysis Report, June 2003.

14. ASME Boiler and Pressure Vessel Code, Appendix K, Section XI, 2001 Edition.

15. EPRI NP-719-SR, T.U. Marston, Flaw Evaluation Procedures: ASME Section XI, Electric Power Research Institute, Palo Alto, California, August 1978.

11.0 Certification

This report is an accurate description of the low upper-shelf toughness fracture mechanics analysis performed for the reactor vessels at Point Beach.

10/15/04

H. P. Gunawardane, Engineer III — Date
Materials and Structural Analysis Unit

This report has been reviewed and found to be an accurate description of the low upper-shelf toughness fracture mechanics analysis performed for the reactor vessels at Point Beach.

10/15/04

A. D. Nana, Principal Engineer — Date
Materials and Structural Analysis Unit

Verification of independent review.

10/15/04

A. D. McKim, Manager — Date
Materials and Structural Analysis Unit

This report is approved for release.

for R.E. Austin 10/15/04

R. E. Austin, Project Development Manager — Date

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12.0 Appendix A

The following pages contain input information from Nuclear Management Company.

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NPL 2004-0139

June 29, 2004

Heshan Gunawardane
AREVA / Framatome ANP, Inc.
MS OF50
3315 Old Forest Road
Lynchburg, VA 24501

Heshan:

This correspondence will serve to formally document the requested inputs for the PBNP Units 1 and 2 RPV Equivalent Margins Assessment that is being performed in accordance with AREVA Proposal FANP-04-1067, April 2, 2004.

Applicable ASME Section XI Code

The PBNP ISI Program is in the fourth ten-year interval, which began on July 1, 2002 for both PBNP-1 and PBNP-2. The program is in accordance with the 1998 edition through 2000 addenda (98A00) of ASME Section XI Code as modified by 10 CFR 50.55a and approved relief requests and code cases. (Reference 1)

Fluence Projections

For the case of full uprated power condition (1678 MWt), without hafnium absorber assemblies, for EOLE (53 EFPY) use the older calculated fluence projections contained in Section 2 of Reference 2. This is requested for input consistency with the remaining RV embrittlement analyses.

For the cases of mini uprated power condition (1540 MWt), with and without hafnium absorber assemblies, for EOLE (53 EFPY) use the revised calculated fluence projections contained in Section 2 of Reference 3.

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Normal Heatup and Cooldown Rates

The PBNP RCS heatup and cooldown rates for normal operation are 100 degrees Fahrenheit per hour for both heatups and cooldowns. (Reference 4)

Predicted Operating Temperatures

The analyses for current licensed rated power conditions (1540 MWt) include a range of full load T(avg)'s from 558.1 to 574 degrees Fahrenheit. The resulting T(hot) and T(cold) ranges are 588.1 to 603.5, and 528 to 544.5 degrees Fahrenheit, respectively (Reference 5). PBNP currently uses a T(avg) program of 547 to 570 degrees Fahrenheit (no load to full load) (Reference 6), resulting in a T(hot) and T(cold) of approximately 597 and 542 degrees Fahrenheit, respectively (Reference 7).

The analyses for the 10.5 percent uprated power condition (1678 MWt) include a range of T(avg) from 558.6 to 573.4 degrees Fahrenheit. The resulting T(hot) and T(cold) ranges are 591.2 to 605.5, and 526 to 541.4 degrees Fahrenheit, respectively (Reference 8).

Transient Information

The original component transients are defined in each RPV design specification (References 9 and 10 for Units 1 and 2, respectively). A revised set of component design transients was generated to support steam generator replacement, a partial power uprate (8.7 percent), and license renewal (Reference 11). The RPV transients were evaluated and characterized for the partial power uprated condition in Reference 12. The RPV transients were further evaluated and characterized for full uprated conditions in Reference 13.

In addition, Chapter 14 of the PBNP FSAR (Reference 14) has been provided via previous correspondence. Chapter 14 contains the PBNP safety analysis summaries. These transients should be reviewed for bounding conditions with respect to the component design transients.

Applicable ASME Section II and III Code

ASME Boiler and Pressure Vessel Code, Section II, 1989, no Addenda.

ASME Boiler and Pressure Vessel Code, Section III, 1989, no Addenda.

AREVA

Sincerely,

WB Fromm

Brad Fromm
PBNP License Renewal
Nuclear Management Company

James E. Knorr

James E. Knorr
Manager of License Renewal PBNP
Nuclear Management Company

bms

References:

1. SER 2001-0010, "Point Beach Nuclear Plant, Units 1 and 2 – Relief Requests RR 1-24 (Unit 1) And RR-2-30 (Unit 2) Re: Use Of ASME Code Section XI, 1998 Edition With Addenda Through 2000 (TAC Nos. MB2230 And MB2231)", dated November 6, 2001.

2. Westinghouse Letter Report, LTR-REA-02-23, "Pressure Vessel Neutron Exposure Evaluations, Point Beach Units 1 and 2, S. L. Anderson, Radiation Engineering and Analysis, February 2002.

3. Westinghouse Letter Report, LTR-REA-04-64, "Pressure Vessel Neutron Exposure Evaluations, Point Beach Units 1 and 2, S. L. Anderson, Radiation Engineering and Analysis, June 2004.

4. Point Beach Nuclear Plant Technical Requirements Manual Pressure Temperature Limits Report, Section 2.1, "RCS Pressure and Temperature Limits (LCO 3.4.3)", page 2.2-2, Revision 1, dated December 20, 2002.

5. NMC Letter, NRC 2002-0075, "Responses to Requests for Additional Information, License Amendment Request 226, Measurement Uncertainty Recapture Power Uprate", August 29, 2002.

6. Setpoint Document, STPT 5.1, "Primary Control Systems Rod Speed Control", Revision 7.

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7. Internal PBNP email, Steve Barkhahn to Brad Fromm, dated 4/17/04.

8. Westinghouse, Power Uprate Project, PBNP Units 1 and 2, Volume 1 NSSS Engineering Report, and Volume 2 BOP Engineering Report, April 2002.

9. Section 4 of Westinghouse Equipment Specification G - 676243, "Reactor Coolant System – Reactor Vessel", Revision 0, 05/05/1966.

10. Section 4, and Figures 1 through 15 of Westinghouse Equipment Specification E-spec 677456, "Addendum to Equipment Specification 676413 Rev. 1, Reactor Coolant System – Reactor Vessel", Revision 2, 07/06/1971.

11. Appendix A of Westinghouse Design Specification, 414A83, "Point Beach Nuclear Plants Units 1 and 2, replacement Reactor Vessel Closure Head (RRVCH)", Revision 0.

12. Appendix B of WCAP-14448, "Addendum to the Stress Reports for the Point Beach Unit Nos. 1 and 2 Reactor Vessels (RSG/Uprating Evaluation), August 1995.

13. Section 5.1.4 of Westinghouse Report, "Power Uprate Project, Point Beach Nuclear Plant, Units 1 and 2, NSSS Engineering Report", April 2002.

14. Chapter 14 of the PBNP Units 1 and 2 Final Safety Analysis Report, June, 2003.

<u>Notes</u>:

References 1, 4, 5, 6, 7, 8, and 14 document the sources of the information.

References 2, 3, 9, 10, 11, 12, and 13 are enclosed.

References 9, 10, 11, 12, and 13 are Westinghouse Proprietary and shall be treated in accordance with the associated Westinghouse Proprietary Agreement established between AREVA/Framatome-ANP, NMC, and Westinghouse in June 2004.

13.0 Appendix B

The following page contains input information from Nuclear Management Company.

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NPL 2004-0236

October 14, 2004

Heshan Gunawardane
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Heshan:

Subject: PBNP Units 1 and 2 Equivalent Margins Assessment Revision, Framatome ANP, Inc. Proposal Number 416 0645, Addendum No. 1

This correspondence will serve to formally document NMC's request to revise the PBNP Units 1 and 2 RPV Equivalent Margins Assessment, Frametome ANP, Inc. Calculation Numbers 77-2647-00 and 77—2647NP-00, to use the 2004 Westinghouse fluence projection as the input to Evaluation Condition 1. Evaluation Condition 1 is full uprated power (1678 MWt), without the presence of Hafnium power suppression inserts.

Sincerely,

WB Fromm

Brad Fromm
PBNP License Renewal
Nuclear Management Company

John G. Thorgersen

John G. Thorgersen for James E. Knorr
Manager of License Renewal PBNP
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